7 CASE STUDIES

7.1 Introduction

This chapter presents case studies providing examples of applications of the information in the Multi-Agency Radiation Survey and Assessment of Materials and Equipment (MARSAME) supplement to the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM). The purpose of these case studies is to illustrate applications of the information in conditions that are frequently encountered and cover a broad range of situations. The general format for each case study mirrors as closely as possible the information presented in MARSAME. References to information, tables, figures, and equations from Chapter 2 through Chapter 6 are provided throughout the case studies.

The MARSAME document contains both procedural as well as informative sections. The case studies provide a practical use of the MARSAME process and as such generally apply only the procedural sections. In addition, much of the information in MARSAME is designed to be applied iteratively. In some case studies, the information will be applied in a different sequence than it is presented in MARSAME because of this iterative nature.

Section 7.2 provides an example of a survey plan for operations within a radiological control area (RCA) at a nuclear power plant. This survey plan provides the user with a starting point for disposition surveys pertaining to materials and equipment (M&E) used within the RCA, and assists the user in selecting the appropriate standard operating procedure (SOP) to complete the disposition survey for each different variety of M&E used. In real operations one SOP often interlocks with other SOPs. In this example, other hypothetical SOPs and their attachments are only referred to by name. They are not explicitly presented in MARSAME.

Section 7.3 provides an example of a disposition survey for a large quantity of bulk material at a mineral processing facility. This represents a special case survey design, establishing gross activity action levels based on normalized effective dose equivalents. These action levels are applied with multiple decision rules using a MARSSIM-type survey design to collect scan survey data as well as systematic and judgmental samples for laboratory analysis.

Section 7.4 and Section 7.5 are based on the same mineral processing facility that serves as the basis for Section 7.3. Section 7.4 provides an example of an interdiction survey for rented heavy equipment that is designed to establish a "baseline" estimate of the residual radioactivity associated with a front loader before it is brought into an RCA for the impacted bulk material. This baseline survey establishes zero net activity as the LBGR and applies MARSAME processes to a Scenario B survey design.

Section 7.5 demonstrates the clearance of the same rented front loader that was brought on to the site in Section 7.4. Section 7.5 describes a Scenario A clearance survey based on the same surface activity action levels to clear the front loader. Sections that contain redundant information have been presented in Section 7.4 only and have been omitted from Section 7.5.

7.2 Release of M&E and Tools from Radiological Control Areas

7.2.1 Description

A work crew has just completed performing a maintenance task inside a radiological control area (RCA) at an operating nuclear facility. Disposition decisions need to be made regarding the M&E used within the RCA.

7.2.2 Objectives

The objective is to make an appropriate disposition decision regarding all of the M&E associated with the maintenance work performed in the RCA. This case study provides an example of applying an existing survey design to a repetitive task.

7.2.3 Initial Assessment of M&E

7.2.3.1 Categorize the M&E

Tools, parts, and other materials must be categorized as impacted or non-impacted prior to exiting the RCA. The existing information is adequate to categorize the M&E (see Figure 2.1). The M&E have been inside an RCA, so all the M&E are impacted. Additional investigations are necessary before a disposition decision regarding the M&E can be made.

7.2.3.2 Design and Implement Preliminary Surveys

Following categorization, the M&E are evaluated to determine whether preliminary surveys are necessary to provide information for designing a disposition survey. The existing information is adequate for selecting a disposition option for the M&E (see Figure 2.2). In addition, the M&E appear to meet the requirements of existing SOPs for releasing M&E from an RCA. No preliminary surveys are necessary.

7.2.3.3 Select a Disposition Option

The selected disposition option depends on the expected future use of the M&E. Tools that will be used in an RCA in the future will be evaluated for reuse in a controlled environment (Option 1, Section 2.4). Other M&E (i.e., other tools, parts, and materials) will be evaluated for release without radiological controls (i.e., clearance, Option 2, Section 2.4).

7.2.3.4 Document the Results of the IA

The documentation requirements for the IA are described in the SOPs. Individual records are not required for each item leaving the RCA. Training records and work schedules show that personnel on duty are properly trained in implementing the SOPs. Quality assurance and quality control (QA/QC) records show that the SOPs are implemented correctly using instruments that are operating properly.

7.2.4 Implement the Survey Design

Since approved SOPs are available for evaluating the M&E, the information on developing a disposition survey design (Chapter 3 and Chapter 4) is not used. This information was used initially to develop the SOPs. Implementation starts with identification of the proper SOP for evaluating specific items leaving the RCA. The M&E are compared to the scope for the SOP to determine whether the SOP is applicable. Then, the M&E are segregated based on which SOP will be applied. The SOP identifies the:

- Description of M&E to which the SOP applies,
- Action level,
- Classification of M&E,
- Number, type, and location of measurements,
- Measurement method (including estimates for uncertainty, detectability, and quantifiability), and
- Documentation requirements for measurement results.

7.2.4.1 Select an Appropriate SOP

All M&E used within the RCA receive a standardized initial assessment and may be surveyed using an existing SOP. The facility has developed and maintains a collection of SOPs providing disposition survey designs for the majority of M&E associated with the facility. The M&E are compared to the description of applicable M&E in the existing SOPs. If no appropriate SOP is available, radiological control of the M&E is maintained.

Each SOP contains the appropriate inputs to the decision and survey design components to reflect the physical and radiological attributes of the appropriate M&E group. Each SOP also contains the action levels, DQOs, MQOs, and QC requirements to validate the quality of measurement data collected using the survey instruments.

The process for returning tools to the tool crib is described in SOP #147, *Maintenance of Tools in Radiological Control Areas*. Tools are cleaned to remove all visible dirt and placed in plastic bags prior to return to the tool crib.

The process for releasing M&E from the RCA depends on the type of M&E being evaluated. The release of hand tools is described in SOP #123, *Clearance of Tools, Materials, and Equipment from Radiological Control Areas*. SOP #123 applies to small items such as hand tools that are being removed from radiological control areas. The assumptions used to develop the detection limits (see Attachment A to SOP #123) and action levels describe the types of M&E where this SOP can be applied. The measurements in this SOP assume activity is distributed on the readily accessible surfaces of surveyed items. This SOP does not apply to surveys of personnel or personal affects, porous materials, paper, trash, and electrical instruments (e.g., power tools, computers, PDAs). Items such as briefcases, pens, papers, personal clothing, etc., are exempt from the release survey requirements of this procedure, unless deemed appropriate by the health physics technician. Exempt items will undergo the same exterior surfaces scanning procedure used in performing a whole body frisk when leaving radiological control areas (see SOP #111).

The release of trash and waste materials is described in SOP #128, *Clearance of Dry Active Waste from Radiological Control Areas*. SOP #128 applies to porous material, paper, and trash being removed from radiological control areas. The assumptions used to develop the detection limits (see Attachment A to SOP #128) and action levels describe the types of M&E where this SOP can be applied. The measurements in this SOP assume uniform activity is distributed volumetrically within surveyed items.

Any questions regarding applicability of an SOP to a specific item are directed to the Radiation Safety Officer or duly authorized representative prior to performing a release survey. Items with inaccessible surfaces are disassembled as completely as possible to thoroughly characterize component materials and equipment in order to facilitate proper release surveys. Items with inaccessible surfaces will not be unconditionally released unless evaluated by designated personnel who authorize and document the release. The evaluation procedure encompasses a review of the history of the item under scrutiny, the radiological conditions of the area in which the item had been used or stored, and the release survey performed.

7.2.4.2 Segregate the M&E

Based on the physical and radiological attributes of the M&E (see Section 2.3), the M&E are segregated based on which SOP will be implemented. For example, tools from the tool crib will be segregated for analysis using SOP #147, *Maintenance of Tools in Radiological Control Areas*, while other tools and parts will be analyzed using SOP #123, *Clearance of Tools, Materials, and Equipment from Radiological Control Areas*.

7.2.4.3 Perform Measurements and Report the Results

Once the M&E are segregated, the measurements described in the SOPs are performed. There is no requirement for documenting individual measurement results. 100% of all M&E leaving the RCA are measured as described in the appropriate SOP.

7.2.5 Assess the Results of the Disposition Survey

Assessment of the disposition survey results is performed while the data are collected. The SOPs include scan-only or in situ designs. Decisions will be made on individual items, so recording individual measurement results is not required (see Figure 6.3). If all of the results are less than the critical value, the M&E demonstrate compliance with the action level. This means that the tools can be returned to the tool crib, or the parts and materials can be cleared for unrestricted use, for example. If any item has a measurement result that exceeds the critical value, additional investigation is required. In most cases tools will be cleaned, while trash or porous items will be evaluated for disposal as low-level radioactive waste.

7.3 Mineral Processing Facility Concrete Rubble

7.3.1 Description

An abandoned mineral processing facility is being redeveloped for commercial/industrial use. The facility processed mineral ores for various metals for over 30 years and was abandoned more than 10 years ago. The processing equipment and existing stockpiles of ore were transferred to another facility when site renovations began. The receiving facility discovered radioactivity levels in excess of background on exterior portions of processing equipment using hand-held Geiger-Mueller (GM) "pancake" detectors.

Prior to discovery of the radioactivity on the processing equipment, the concrete floors had been removed from the processing buildings and stockpiled on-site. Note that if the buildings were still intact, they could be surveyed using a MARRSIM survey. An investigation is performed to trace the source of the radioactivity to the appropriate portion(s) of the mineral processing facility.

7.3.2 Objectives

The objective is to make an appropriate disposition decision regarding the concrete rubble from the impacted portions of the mineral processing facility. It is anticipated that leaks of potentially radioactive processing liquids could have occurred throughout the operating lifetime of the facility. Airborne radioactive concrete dust may have been released during demolition activities, which could have exposed construction personnel and contacted components of the demolition equipment.

7.3.3 Initial Assessment of the M&E

7.3.3.1 Categorize the M&E

As part of the IA, it is necessary to determine whether the concrete rubble is impacted or not. A visual inspection of the concrete rubble was performed. Historical records from the facility concerning sources of ore, ore processing techniques, waste disposal practices, industrial accidents, as well as building and equipment repairs, modifications, and upgrades were reviewed. Interviews with key facility personnel were also performed. In addition, research into mineral processing techniques and radionuclide content of raw ores was performed to obtain additional process knowledge.

Process knowledge indicated the facility processed ilmenite ore (iron titanium oxide, $FeTiO_3$) and produced titanium dioxide. A sentinel measurement of a small amount of ilmenite ore remaining at the site was analyzed by alpha spectrometry and found to contain elevated levels of natural uranium and thorium. Additional measurements performed on the radioactive processing equipment determined that natural uranium and thorium were the source of radioactivity on the processing equipment.

Site history indicates that the general layout of the process was unchanged over the lifetime of the facility, and it is likely that spills occurred repeatedly in discrete locations. Processing liquids and slurries were considered hazardous because of their low pH; radioactivity was not considered an issue. Limited information regarding site history and operations was obtained through interviews with former employees and review of historical documentation. Former employees stated that spills and leaks of process liquids and slurries occurred periodically in several areas of the processing plant; these represent the only potential source of radioactivity in the plant. Fluid spills were quickly corrected by neutralizing the acid to protect employees and equipment. Spills frequently resulted from seal failure within the various pumps in use at the processing operation.

Results from the visual inspection indicated there was a reasonable potential for radioactivity from plant activities to be associated with the concrete rubble. Several chunks of concrete rubble are obviously discolored from plant operations, indicating possible locations of spills. The facility floor consisted of reinforced concrete on a gravel base mat. Portions of the rubble contain possible evidence of staining. The rubble still contains rebar which, for operational reasons, must be segregated and treated as a separate waste stream.

The concrete rubble is considered to be impacted due to the discovery of residual radioactivity on exterior portions of the processing equipment, historical records that acidic process fluids may have spilled on the concrete floor, and process knowledge that the acidic process fluids were mixed with raw ore containing elevated levels of naturally-occurring radioactive material

(NORM) from the uranium and thorium radioactive decay series. The results of the sentinel measurement performed on the raw ore support the categorization as impacted.

7.3.3.2 Design and Implement Preliminary Surveys

Table 7.1 lists the physical attributes of the concrete rubble. No data gaps associated with the physical attributes were identified.

Table 7.1 Physical Attributes of the Concrete Rubble

Attribute	Description
Dimensions	Total Mass 400 ft × 100 ft × 1 ft ≈ 40,000 ft^3 40,000 ft^3 × 0.0283 m^3/ft^3 ≈ 1,132 m^3 The approximate density of crushed concrete is 2.3×10^6 g/m^3 1,132 m^3 × 2.3×10^6 g/m^3 = 2.60×10^9 g = 2.60×10^6 kg Shape The concrete has been broken into chunks less than one meter in the largest dimension. The concrete is stored in three piles, each pile is approximately 1.5 m high, 6 m wide, and 40 m long.
Complexity	Rebar used to reinforce the floor is present in the concrete rubble. The rebar will be segregated and removed, and treated as a separate waste stream.
Accessibility	The concrete rubble may require further reduction in size to ensure measurability.
Inherent Value	The concrete represents inherent value for several potential disposition options. Crushed concrete serves many useful purposes, including recyclable use as roadbed material. This option presents potential cost savings over using virgin materials in place of recycled concrete and a reuse scenario that avoids the relatively high cost for disposal.

Table 7.2 lists the known radiological attributes associated with the concrete rubble, as well as data gaps showing where additional information is required to design a disposition survey. As presented, the existing information is not adequate to design a disposition survey. Preliminary surveys were designed and implemented to address the data gaps identified in Table 7.2. The results of the preliminary surveys were used to modify the conceptual site model by filling some of the data gaps.

Table 7.2 Radiological Attributes of the Concrete Rubble

Attribute	Description			Data Gaps
Radionuclides	Uranium Series Radionuclides	Principal Emission Particle	Emission Energy (MeV)	The radioactivity is likely to have come in contact with the M&E through spills of process fluids and dumping of solid tailings on the concrete floor. Equilibrium status of the decay series is unknown, although sufficient time has elapsed since site closure for the thorium series to have re-established secular equilibrium.
	^{238}U	Alpha	4.20	
	^{234}Th	Beta	0.1886	
	^{234m}Pa	Beta/Gamma	2.28/1.001	
	^{234}Pa	Beta	0.224	
	^{234}U	Alpha	4.77	
	^{230}Th	Alpha	4.688	
	^{226}Ra	Alpha/Gamma	4.78/0.186	
	^{222}Rn	Alpha	5.49	
	^{218}Po	Alpha	6.00	
	^{214}Pb	Beta/Gamma	0.67/0.352	
	^{214}Bi	Beta/Gamma	1.54/0.609	
	^{214}Po	Alpha	7.687	
	^{210}Pb	Beta	0.016	
	^{210}Bi	Beta	1.161	
	^{210}Po	Alpha	5.305	
	Thorium Series Radionuclides	Principal Emission Particle	Emission Energy (MeV)	
	^{232}Th	Alpha	4.01	
	^{228}Ra	Beta	0.0389	
	^{228}Ac	Beta/Gamma	1.17/0.911	
	^{228}Th	Alpha	5.42	
	^{224}Ra	Alpha	5.686	
	^{220}Rn	Alpha	6.288	
	^{216}Po	Alpha	6.78	
	^{212}Pb	Beta/Gamma	0.334/0.238	
	^{212}Bi	Alpha/Beta	6.05/2.246	
	^{212}Po (64%)	Alpha	8.785	
	^{208}Tl (36%)	Beta	1.80	
Activity	Activity levels range from background (approximately 40 Bq/kg) to 4000 Bq/kg from isolated portions of the concrete rubble where spills occurred.			The expected range of activity is an estimate. Nature and extent of activity needs to be investigated to provide better estimates of average and maximum activity. Better estimates of background are needed.
Distribution	The radioactivity is heterogeneously distributed throughout the mass of concrete rubble.			No data gaps were identified. The current distribution is not a concern since the concrete will be crushed to 2-3 cm size prior to survey.
Location	The concrete rubble is considered a volumetrically impacted mass. The residual radioactivity that is present is a combination of fixed and removable.			The distribution of radioactivity with depth may provide useful information for selecting measurement methods because it can impact the total measurement efficiency.

The radionuclides of potential concern are the uranium (^{238}U) and thorium (^{232}Th) natural radioactive decay series. Based on process knowledge, radionuclide concentrations in the raw ore average between 750 and 1,100 Bq/kg for members of the uranium series, and between 200 and 400 Bq/kg for members of the thorium series. Following processing, some ^{238}U and ^{232}Th decay products may not have been in equilibrium with the parents. The amount of time since the plant ceased operations (i.e., 10 years) indicates there is a potential for the thorium series radionuclides to have re-established secular equilibrium. Preliminary survey measurements are required to determine the equilibrium status of the uranium and thorium series radionuclides.

Limited scanning of concrete rubble was performed using a GM pancake detector. The purpose of the scanning was to determine how the radioactivity associated with the concrete was distributed. The scanning survey also included additional visual inspection of the concrete.

Intermittent staining within the concrete rubble and scanning surfaces of concrete chunks demonstrates that the radioactivity was heterogeneously deposited on the processing building floor. Higher levels of radioactivity were found in areas where spills occurred historically (i.e., discolored concrete). The staining did not appear to have penetrated more than one-quarter inch into the concrete when the floor was intact. Prior to demolition, the presence of cracks and other structural irregularities in the concrete floor provided preferential pathways for activity to penetrate to greater depths. This resulted in some variance in activity with depth of the original concrete floor.

Samples were collected from the crushed concrete from the processing mill floor to determine concentrations of residual radioactivity using alpha spectrometry and gamma spectroscopy. Concrete samples were collected from four biased locations, including two areas of elevated gross activity within the concrete rubble with GM readings as high as 250 cpm and visible staining (Samples 1 and 2), and two samples with readings consistent with the average readings observed during scanning (40 to 45 cpm) (Samples 3 and 4). Process knowledge and limited historical site information indicates that radiological materials were never used or stored within the on-site administrative building. Reference Samples 1 and 2 were collected from the concrete floor in this facility to provide information on background activities in non-impacted concrete for the uranium and thorium decay series for the conceptual model. The six samples were sent to a radioanalytical laboratory for analysis, and the results of the analyses are provided in Table 7.3.

Table 7.3 Preliminary Survey Analytical Results

Alpha Spectrometry Results for Uranium Series Radionuclides (Bq/kg)

Sample ID	^{234}U	CSU[1]	MDC[2]	^{235}U	CSU[1]	MDC[2]	^{238}U	CSU[1]	MDC[2]
Sample 1	7,000	± 2,100	1,900	340	± 1,900	1,600	7,600	± 2,400	1,900
Sample 2	7,200	± 2,300	1,900	320	± 1,700	1,600	7,000	± 2,100	1,900
Sample 3	21	± 7.4	3.7	0.74	± 1.9	0.74	21	± 7.0	3.7
Sample 4	25	± 8.1	3.7	0.74	± 3.0	0.74	21	± 7.0	3.7
Reference Sample 1	19	± 5.2	3.7	0.37	± 0.74	0.74	20	± 5.6	3.7
Reference Sample 2	13	± 3.7	3.7	0.37	± 0.74	0.74	11	± 3.3	3.7

Alpha Spectrometry Results for Thorium Series Radionuclides (Bq/kg)

Sample ID	^{232}Th	CSU[1]	MDC[2]	^{228}Th	CSU[1]	MDC[2]
Sample 1	1,400	± 110	110	1,300	± 150	110
Sample 2	1,200	± 130	110	1,500	± 190	110
Sample 3	21	± 1.5	1.1	23	± 1.5	1.1
Sample 4	26	± 1.1	1.1	24	± 1.1	1.1
Reference Sample 1	21	± 1.1	1.1	22	± 1.1	1.1
Reference Sample 2	23	± 1.1	1.1	23	± 1.1	1.1

Gamma Spectroscopy Results for Uranium Series Radionuclides (Bq/kg)

Sample ID	^{214}Bi	CSU[1]	MDC[2]	^{214}Pb	CSU[1]	MDC[2]	^{226}Ra	CSU[1]	MDC[2]
Sample 1	93	± 920	1,400	530	± 780	1,300	47	± 1,100	1,500
Sample 2	740	± 1,000	1,300	1,000	± 870	1,200	192	± 1,200	1,400
Sample 3	21	± 1.1	3.6	21	± 1.1	6.3	64	± 9.6	16
Sample 4	22	± 1.1	4.1	23	± 1.1	7.0	68	± 8.5	19
Reference Sample 1	17	± 1.1	3.1	17	± 1.1	7.0	36	± 6.3	18
Reference Sample 2	20	± 1.1	3.4	20	± 1.1	5.6	52	± 7.1	17

Gamma Spectroscopy Results for Thorium Series Radionuclides (Bq/kg)

Sample ID	^{228}Ac	CSU[1]	MDC[2]
Sample 1	1,600	± 180	52
Sample 2	1,400	± 130	41
Sample 3	14	± 2.6	4.4
Sample 4	21	± 3.1	6.3
Reference Sample 1	15	± 3.3	5.9
Reference Sample 2	16	± 3.4	3.4

[1] CSU is the combined standard uncertainty of the measurement result reported by the analytical laboratory.
[2] MDC is the minimum detectable concentration reported by the analytical laboratory.

7.3.3.3 Select a Disposition Option

The preferred disposition of the concrete rubble is clearance. It is expected that the concrete will be reused as roadbed or disposed of in a municipal landfill. If the activity levels exceed the project action levels, then the concrete may need to be disposed of as discrete naturally-occurring or accelerator-produced (NARM) waste. If the activity is below the alternate action levels, the concrete may either be reused or disposed of as diffuse NARM waste.

7.3.3.4 Document the Results of the IA

The results of the IA were documented in a letter report. The purpose of the letter report was to document the categorization decision and all supporting information. The letter report was reviewed and finalized by the facility owner. Detailed results of the IA will be included in the final documentation of the survey design.

7.3.4 Identify Inputs to the Decision

Following completion of the IA, additional information was needed to develop the disposition survey design.

7.3.4.1 Finalize the List of Radionuclides to be Measured

The list of radionuclides of concern was finalized based on the preliminary survey results. Uranium-238, ^{234}U, and ^{226}Ra are the radionuclides of concern for the uranium natural decay series. The alpha spectrometry results indicate that ^{238}U and ^{234}U are in equilibrium (i.e., have equal concentrations). Since alpha spectrometry for uranium isotopes provides results for both ^{238}U and ^{234}U, both isotopes (and their decay products with half-lives less than six months) will be kept as radionuclides of concern. There is no indication of enrichment or depletion of uranium as a result of site activities based on the uranium alpha spectrometry results listed in Table 7.3.

Radium-226 decay products, including ^{210}Pb, are assumed to be out of secular equilibrium with the other uranium series radionuclides (e.g., ^{238}U and ^{234}U) because process knowledge shows the chemical processing at the plant would separate uranium from radium. Bismuth-214 and ^{214}Pb can be used as beta or gamma emission surrogates for ^{226}Ra, because the decay products of ^{226}Ra should be in secular equilibrium with one another. However, a twenty-one day ingrowth period may be required to confirm this assumption. The planning team determined an ingrowth study was not required for this project following discussions with the regulators.

Thorium-232 is the radionuclide of concern for the thorium natural decay series. Based on the alpha spectrometry and gamma spectroscopy results shown in Table 7.3, all members of the thorium natural decay series are in secular equilibrium. Actinium-228 emits gamma rays that are easy to quantify using gamma spectroscopy, and can be used as a surrogate for the members of the thorium series.

7.3.4.2 Select an Action Level

An action level of 0.01 mSv/y was selected based on discussions with the stakeholders. Using information provided in NUREG-1640 (NRC 2003), the action levels were converted into concentration units based on clearance as the disposition option. Incorporating the concrete rubble into roadbed material would provide the highest potential doses following clearance. The mean values from NUREG-1640 (NRC 2003), Table I 1.13 (Normalized effective dose equivalents from all pathways: Driving on road [μSv/y per Bq/g]) were selected as the basis for the action levels.

Radionuclide of concern	^{238}U	^{234}U	^{232}Th	^{226}Ra
Mass-based EDE mean values (μSv/y per Bq/g)	0.26	8.2×10^{-4}	30	22

The action levels from Table I1.13, NUREG-1640 (NRC 2003) are expressed in units of μSv/y per Bq/g, but the preliminary survey measurement results are in Bq/kg. To make a direct comparison, the action levels were converted to units of Bq/kg. The action levels were converted to concentrations by inverting the action levels and multiplying by the selected dose limit (i.e., the inverted action levels in units of Bq/g per μSv/y are multiplied by 0.01 mSv/y, 1,000 g/kg, and 1,000 μSv/ mSv providing action levels in Bq/kg). Table 7.4 lists the action levels in concentration units of Bq/kg.

Table 7.4 Radionuclide-Specific Action Levels

Radionuclide	Mass-Based EDE Mean Values (Bq/g per μSv/y)	Action Level (Bq/kg)
^{238}U	$\frac{1\,\text{Bq/g}}{0.26\ \mu\text{Sv/y}} \times 0.01\,\text{mSv/y} \times 1 \times 10^{6} = 38{,}000$	38,000
^{234}U	$\frac{1\,\text{Bq/g}}{8.2 \times 10^{-4}\ \mu\text{Sv/y}} \times 0.01\,\text{mSv/y} \times 1 \times 10^{6} = 12{,}000{,}000$	12,000,000
^{232}Th	$\frac{1\,\text{Bq/g}}{3.0 \times 10^{1}\ \mu\text{Sv/y}} \times 0.01\,\text{mSv/y} \times 1 \times 10^{6} = 330$	330
^{226}Ra	$\frac{1\,\text{Bq/g}}{2.2 \times 10^{1}\ \mu\text{Sv/y}} \times 0.01\,\text{mSv/y} \times 1 \times 10^{6} = 450$	450

The unity rule (Equation 7-1) was used to account for the individual radionuclide action levels. The unity rule is satisfied when the summed analyses of each radionuclide against its respective action level yields a value less than one:

$$\text{The Unity Rule} = \frac{C_1}{AL_1} + \frac{C_2}{AL_2} + \ldots \frac{C_n}{AL_n} \leq 1 \qquad \textbf{(7-1)}$$

Where:

C = Concentration of each individual radionuclide (1, 2, … n)

AL = Action level value for each individual radionuclide (1, 2, … n)

Equation 7-1 was used to calculate the sum of fractions for each of the preliminary survey results:

$$\text{The Unity Rule} = \frac{C_{^{238}\text{U}}}{AL_{^{238}\text{U}}} + \frac{C_{^{234}\text{U}}}{AL_{^{234}\text{U}}} + \frac{C_{^{232}\text{Th}}}{AL_{^{232}\text{Th}}} + \frac{C_{^{226}\text{Ra}}}{AL_{^{226}\text{Ra}}} \leq 1$$

$$\text{Sample 1} = \frac{7{,}600 \text{ Bq/kg}}{38{,}000 \text{ Bq/kg}} + \frac{7{,}000 \text{ Bq/kg}}{12{,}000{,}000 \text{ Bq/kg}} + \frac{1{,}400 \text{ Bq/kg}}{330 \text{ Bq/kg}} + \frac{47 \text{ Bq/kg}}{450 \text{ Bq/kg}} = 4.5$$

$$\text{Sample 2} = \frac{6{,}900 \text{ Bq/kg}}{38{,}000 \text{ Bq/kg}} + \frac{7{,}200 \text{ Bq/g}}{12{,}000{,}000 \text{ Bq/kg}} + \frac{1{,}230 \text{ Bq/kg}}{330 \text{ Bq/kg}} + \frac{192 \text{ Bq/kg}}{450 \text{ Bq/g}} = 4.2$$

$$\text{Sample 3} = \frac{21 \text{ Bq/kg}}{38{,}000 \text{ Bq/kg}} + \frac{21 \text{ Bq/kg}}{12{,}000{,}000 \text{ Bq/kg}} + \frac{21 \text{ Bq/kg}}{330 \text{ Bq/kg}} + \frac{64 \text{ Bq/kg}}{450 \text{ Bq/g}} = 0.21$$

$$\text{Sample 4} = \frac{21 \text{ Bq/kg}}{38{,}000 \text{ Bq/kg}} + \frac{25 \text{ Bq/kg}}{12{,}000{,}000 \text{ Bq/kg}} + \frac{26 \text{ Bq/kg}}{330 \text{ Bq/kg}} + \frac{68 \text{ Bq/kg}}{450 \text{ Bq/g}} = 0.23$$

The results of the calculations for Samples 1 and 2 exceed a sum of fractions of 1.0, and indicate the presence of small volumes of concrete with elevated activity. Note that the reported MDCs for gamma spectroscopy for ^{226}Ra in Samples 1 and 2 would not meet the MQOs for clearance (i.e., the MDC exceeds the action level). However, the radionuclide concentrations in these two samples clearly exceed the action level. Therefore, the quality of these results is acceptable to support the disposition survey design.

The results of the calculations for Samples 3 and 4 indicate that, on average, the concrete rubble is expected to have radionuclide concentrations below the action levels. Therefore, the average activity in the concrete rubble is expected to be below the action level. Large blocks containing elevated levels of radioactivity may be visually identified via staining, verified with a GM detector, and segregated prior to removal of the rebar.

7.3.4.3 Identify the Parameter of Interest

Since the disposition option is stated in terms of dose, the parameter of interest is the mean radionuclide concentration. The target population is all of the possible measurement results that could be obtained within a survey unit. This means the target population will be defined by the survey unit boundaries (Section 7.3.4.4) and the selected measurement method (Section 7.3.4.5).

7.3.4.4 Survey Unit Boundaries

Survey unit boundaries are based primarily on the modeling assumptions used to develop the action levels. The volume of concrete used to model exposures for building a road is 83 cubic meters (NUREG-1640 (NRC 2003) Volume 2, Appendix B, Tables B-8 and B-11). Each survey unit will consist of approximately 80 cubic meters of crushed concrete (approximately 25 meters × 22 meters × 0.15 meters).

The volume of concrete poured to create the floor of the processing mill was approximately 1,100 cubic meters. Crushing the concrete and removing the rebar is expected to result in approximately a 25% increase in volume due to air gaps, for a total volume of 1,400 cubic meters of crushed concrete. Using these calculations, there will therefore be a total of 18 survey units plus one reference area.

The concrete rubble can be spread into a relatively uniform layer approximately 15 centimeters thick and scanned. This adapts an approach used in MARSSIM to survey the top 15 centimeters of surface soil as a two-dimensional object.

7.3.4.5 Inputs for Selection of Measurement Methods

The selected measurement method will be required, at a minimum, to detect radionuclide concentrations at or below the action levels in Table 7.4. The survey planners considered each of the three possible measurement techniques (see Section 5.9.1).

Scan-only techniques have the ability to detect surface activity at concentrations below the action levels, as shown in Appendix F. In situ measurement techniques are also expected to have the ability to measure radionuclide concentrations at the action levels. However, uncertainties associated with the efficiency for both techniques will be large. In order to reduce these uncertainties to a level where the radionuclide concentrations are measurable, the concrete would need to be pulverized and mixed rather than just crushed to 2-3 cm size. Since the cost of processing the concrete this way would be a major cost associated with the disposition survey, a MARSSIM-type survey design was selected for the disposition survey.

Concrete samples will be analyzed in a laboratory using alpha spectrometry for uranium isotopes (i.e., ^{234}U and ^{238}U) as well as gamma spectroscopy for other radionuclides of concern (i.e., ^{214}Bi, ^{214}Pb, and ^{228}Ac). Sample sizes must be sufficient to allow quantification of radionuclide concentrations at the action levels. By convention, the MQC for each radionuclide of concern is selected so the measurement method uncertainty at concentrations equal to the action levels in Table 7.4 is 10%. Alternatively, the samples can be sealed in an airtight container for twenty-one days to allow secular equilibrium to be reestablished.

Due to the rough, irregular shape of the concrete rubble, alpha radiation is easily attenuated and is difficult to measure. Beta and gamma measurements typically provide a more accurate assessment of thorium and uranium activity on most building surfaces because surface conditions cause significantly less attenuation of beta and gamma particles than alpha particles. For this reason, scanning will be performed using instruments that detect beta or gamma radiation. Surface scans, using a 12.7-centimeter by 0.16-centimeter field instrument for detection of low-energy radiation (FIDLER) sodium iodide (NaI[Tl]) scintillation probe, are used to scan for gamma emissions. The approximate detection sensitivity of the FIDLER is 300 Bq/kg for natural uranium and 20 Bq/kg for natural thorium (see Appendix F) when activity is present at the surface. The FIDLER has a large probe and can detect gammas from a greater height above the crushed concrete than alpha or beta detection equipment, making it a more practical choice for surveying large volumes of material. The selection of the FIDLER over more conventional NaI(Tl) detectors (e.g., a three-inch by three-inch gamma scintillation detector) is primarily based on the FIDLER's ability to detect low-energy gamma radiation, which comprises the majority of the gamma radiation from the radionuclides of concern.

7.3.4.6 Modify the Action Levels to Account for Multiple Radionuclides

Radionuclide-specific action levels need to be combined into a single gross gamma action level for evaluating the FIDLER scan measurements. The information in Section 3.3.3.1 requires an estimate of the relative fraction of the total activity contributed by each radionuclide. A consistent relationship between ^{238}U and ^{232}Th concentrations is not expected based on the IA, since different ore bodies could contain different ratios of these radionuclides. Rather than develop a preliminary survey attempting to develop this relationship, a conservative approach was adopted for this project.

Assuming the entire radioactivity detected by the FIDLER results from the presence of the most restrictive radionuclide will provide the most conservative gross gamma action level. The ratios of exposure rate to radionuclide concentration (μR/h per Bq/kg) and instrument response to exposure rate (cpm per μR/h) were developed in Appendix F during development of the scan MDC for both ^{238}U and ^{232}Th. These ratios can be used to calculate the count rate above background associated with a radionuclide activity equal to the action level as shown in Equation 7-2.

$$GG_{AL} = AL \times \left(\frac{\mu\text{R/h}}{\text{Bq/kg}} \right) \times \left(\frac{\text{cpm}}{\mu\text{R/h}} \right) \tag{7-2}$$

Where:

GG_{AL} = Gross gamma action level (cpm)

AL = Action level value for each individual radionuclide (Bq/kg)

Equation 7-2 was used to calculate a gross gamma count rate above background for the FIDLER assuming each radionuclide of concern was present at a concentration equal to the action level. The gross gamma count rates were divided by two to account for uncertainty associated with the detector efficiency calculation (see Appendix F) and added to the background count rate from Appendix F. The result is a gross gamma action level for the FIDLER to identify locations with unexpectedly high gamma activity that could result in doses near the action level of 0.01 mSv/y. The results of the calculations are shown in Table 7.5. The ^{232}Th gross gamma action level of 30,000 cpm is more conservative than the ^{238}U gross gamma action level of 140,000 cpm, so 30,000 cpm was selected as the gross gamma action level.

FIDLER readings that exceed the ^{232}Th gross gamma action level indicate locations where radionuclide concentrations could result in doses exceeding 0.01 mSv/y if all of the activity results from ^{232}Th.

Table 7.5 Calculation of the Gross Gamma Action Level

Action Level (Bq/kg)	μR/h per Bq/kg (from Appendix F)	cpm per μR/h (from Appendix F)	Gross Gamma Count Rate (cpm)	Adjusted Gross Gamma Count Rate (cpm)	Background Count Rate (cpm)	Gross Gamma Action Level (cpm)
^{238}U 38,000	1.413E-04	45,593	244,807	122,404	12,870	140,000
^{232}Th 330	2.619E-02	3,923	33,905	16,953	12,870	30,000

Since ^{232}Th has decay products in secular equilibrium that can be used to estimate the ^{232}Th activity, gamma spectroscopy can be used to quantify ^{232}Th concentrations. FIDLER readings that exceed 140,000 cpm identify locations where radionuclide concentrations could result in doses exceeding 0.01 mSv/y if all of the activity results from ^{238}U. Alpha spectrometry is required to quantify ^{238}U concentrations.

7.3.4.7 Identify Alternative Actions

The alternative actions identify the results of decisions based on the measurement results. If the radionuclide concentrations do not result in a dose that exceeds the action level, the material is cleared. If the dose exceeds the action level, materials exceeding the action level will be segregated and investigated for disposal as NARM waste.

7.3.4.8 Decision Rules

MARSSIM-type surveys are designed to evaluate the average radionuclide concentration in a survey unit using samples or direct measurements, as well as small areas of elevated activity using scans. Small areas of elevated activity receive additional investigation. Since there are multiple action levels and multiple decisions to be made, there are multiple decision rules for the disposition survey. The first two decision rules address how small areas of elevated activity are identified by scans and what investigations will be performed. The third decision rule evaluates the results of the investigations of small areas of elevated activity. The fourth decision rule evaluates the average activity in each survey unit.

1. If any FIDLER scanning measurement result exceeds the gross gamma action level of 30,000 cpm (see Section 7.3.5.4), a biased sample will be collected for laboratory analysis by gamma spectroscopy, otherwise no biased samples will be collected.
2. If any FIDLER scanning measurement exceeds 140,000 cpm, the biased sample collected for gamma spectroscopy analysis will also be analyzed by alpha spectrometry for uranium and thorium isotopes, otherwise the concrete will be held awaiting the results of the gamma spectrometry analysis.
3. If the results from a biased sample result in a sum of fractions for ^{238}U, ^{234}U, ^{226}Ra, and ^{232}Th exceeding 1.0, the concrete will be segregated and investigated for disposal as NARM waste. Otherwise, the survey unit will be evaluated based on the WRS test results for the samples taken over a systematic grid.
4. If the mean sum of fractions in a survey unit exceeds 1.0, the concrete will be segregated and investigated for disposal as NARM waste. Otherwise, the WRS test will be performed to support the final disposition decision for that survey unit.

7.3.4.9 Reference Materials

Concrete from the administrative building contains non-impacted materials, as established by the process knowledge discussed in Section 7.3.3.1. The reference material measurements will be performed on the floor in the administrative building. The geometry of the floor is similar enough to the concrete rubble (after crushing to 2-3 cm size and arrangement into a 15 cm thick layer) that modifications to the building are not required.

7.3.5 Survey Design

The concrete rubble from the mineral processing facility is surveyed for clearance using a MARSSIM-type disposition survey. The survey includes scanning to identify small areas of elevated activity combined with collection and analysis of samples to evaluate the average activity in the concrete rubble.

Scenario A will be used to design the survey, since decisions will be made based on average radionuclide concentrations and radioactivity levels in each survey unit. The null hypothesis is that the radionuclide concentrations in the concrete rubble will result in a dose that exceeds 0.01 mSv/y. There are two decisions for MARSSIM-type surveys. The first decision is based on the average radionuclide concentrations in the survey unit, and the second decision is based on the scanning survey results and subsequent biased sample results from flagged locations. The same null hypothesis applies to both decisions.

A Type I decision error would occur if the decision maker decided the activity levels in the concrete rubble were below the action level when they actually exceeded the action level. The consequences of making this decision error could result in increased doses to members of the public and failing to identify small areas of elevated radionuclide concentrations. The stakeholders agreed to a Type I decision error rate of 5% based on the consequences of making this decision error. This Type 1 error rate applies to both the scanning portion of the survey design as well as sampling on a systematic grid.

A Type II decision error would occur if the decision maker decided the activity levels in the concrete rubble exceeded the action level when they were actually below the action level. The consequences of making this decision error could result in increased disposal costs. The stakeholders agreed to a Type II decision error rate of 10% based on the consequences of making this decision error for sampling. However, during scanning the consequences of making this decision error are simply collecting additional data, so a Type II decision error rate of 60% is selected for the scanning surveys (i.e., deciding to stop and count longer when no radioactivity is present).

7.3.5.1 Classification

All of the concrete rubble from the floor of the processing facility has the potential to exceed one or more of the action levels. The concrete rubble is classified as Class 1 M&E.

7.3.5.2 Scanning Survey Design

The concrete must be crushed prior to performing the scanning survey to reduce the size of individual particles to less than 2-3 cm in diameter. This provides a uniform matrix of material ensuring a representative sample can be collected, and also allows the rebar to be removed. The crushed concrete is distributed in a layer approximately 15 cm thick, and surveyed using a FIDLER at a height of 10 cm above the surface. The scan speed is 0.25 meter per second, which is consistent with the scan MDC calculations (see Appendix F). One hundred percent of the concrete rubble is scanned with readings in excess of 30,000 cpm flagged for additional investigation. The additional investigations include collection and analysis of samples using gamma spectrometry to quantify activity levels for the radionuclides of concern. Samples collected from locations with readings in excess of 140,000 cpm are also analyzed for uranium and thorium isotopes by alpha spectrometry.

7.3.5.3 Sample Collection Survey Design

The concrete rubble is divided into survey units and a statistically based number of samples are collected from each survey unit. Since multiple radionuclides are present, the unity rule is used to evaluate the sample results. Since the radionuclides are present in background, the Wilcoxon Rank Sum (WRS) test is used to evaluate the survey results.

The upper bound of the gray region (UBGR) is set equal to the action level, which is a sum of fractions of 1.0 above background. The lower bound of the gray region (LBGR) is set equal to the expected sum of fractions based on results from the preliminary survey. The expected average activity in the concrete rubble is close to background, even though isolated areas have results more than four times the action level. An LBGR value of 0.15 is selected, which is consistent with results reported in Table 7.3 for the two randomly selected samples (i.e., Samples 3 and 4). Since the values are not corrected for background, this value is considered conservative. The shift (UBGR - LBGR) is 0.85.

The variability in the activity levels for the concrete rubble is not well defined. To be conservative, the variability in the results should be large for results near the LBGR. A value of 0.15 was selected for the variability. This value is equal to the LBGR, and represents 100% variability in results that are at or near background. The relative shift equals 5.6 (0.85 divided by 0.15 and rounded down). Since relative shifts greater than 4.0 do not result in significantly smaller numbers of samples, a relative shift of 4.0 was used to determine the number of samples and also help to ensure adequate statistical power.

Table A.2b (Appendix A) lists the number of samples required for each survey unit and reference area for use with the WRS test. Seven samples are required for each survey unit and reference area using a relative shift of 4.0, Type I decision error rate of five percent, and Type II decision error rate of 10 percent. The radionuclide or radioactivity concentrations derived from the dose-based action level are based on an average radionuclide concentration or level of radioactivity over the entire survey unit. No adjustments need to be made to the number of measurements to account for the scan MDC, since the scan MDC is less than the action level for both ^{238}U and ^{232}Th.

Seven samples of approximately 1,000 grams of concrete rubble are collected from each survey unit. This mass corresponds to a cylinder with a diameter of approximately 6 cm (2.5 inches) to a depth of 15 cm (6 inches). This disposition survey design will be applied to all of the concrete rubble, including the concrete segregated based on visual inspection and elevated scanning results with a GM detector during the preliminary surveys (see Section 7.3.4.2).

7.3.5.4 Develop an Operational Decision Rule

The action level is stated in terms of incremental dose above background. In a MARSSIM survey, there are requirements for both sample measurements and scanning results. Samples will be collected from non-impacted concrete to represent background radionuclide concentrations. The WRS test will be used to evaluate the survey results. If the test statistic for the WRS test is less than or equal to 65 ($n = m = 7$, $\alpha = 0.05$), decide that the dose from that survey unit exceeds 0.01 mSv/y and the concrete will not be cleared.

For the scanning results, if any FIDLER measurement exceeds 30,000 cpm, collect a biased concrete sample at the location of the elevated measurement for analysis by gamma spectroscopy. If any FIDLER measurement exceeds 140,000 cpm, analyze the biased concrete sample by alpha spectrometry as well. If the sum of fractions for any biased sample exceeds 1.0, decide that the dose from that survey unit exceeds 0.01 mSv/y and the concrete will not be cleared.

7.3.5.5 Document the Survey Design

The final survey design was documented in a detailed work plan. The work plan provided the results of the IA, as well as all of the assumptions used to develop the survey design. The DQOs and MQOs for the survey design were also included.

The draft work plan was submitted to the stakeholders for review. Comments were received, and responses to comments developed and approved. The approved responses to comments were incorporated into a final work plan documenting the disposition survey design.

7.3.6 Implement the Disposition Survey Design

7.3.6.1 Protection of Health and Safety

A job safety analysis (JSA) was performed based on the tasks defined in the work plan documenting the disposition survey design. Table 7.6 shows the results of the JSA. Potential health and safety hazards identified by the JSA are addressed in a site-specific health and safety plan. No hazards associated with the concrete rubble will notably affect how the disposition survey is implemented.

7.3.6.2 Segregation

Concrete rubble with visible stains and pitting on the floor surface is segregated as having higher activity concentrations. Stained and unstained concrete were grouped into separate survey units. Following segregation, the concrete was crushed to 2-3 cm diameter pieces and the rebar was removed.

Table 7.6 Job Safety Analysis for Surveying Concrete Rubble

Sequence of Basic Job Steps	Potential Hazards	Recommended Action or Procedure
1. Dividing rubble into manageable survey units	Use of front end loader by untrained personnel	Ensure equipment operators are adequately trained
	Personnel in area could be struck by heavy equipment	Area workers must maintain eye contact with equipment operators
		Reflective vests will be worn to improve visibility
	Exposure to silica	Use of a real-time dust monitor will document dust levels. Respiratory protection will be used if dust levels exceed established action levels (dependent on silica content of concrete)
	Lower back strain from lifting	Proper lifting techniques will be used
		Loads will be sized so as not to create unreasonable weights for manual lifting
	Exposure to radiological contamination	PPE including booties, Tyveks, and gloves will be used
2. Establish exclusion zone for survey area	None anticipated	
3. Use hand-held survey instruments to perform survey measurements on the crushed concrete	Unstable footing may result in slips, trips, or falls	Spread out rubble in a way to minimize tripping hazards by creating clear rows between rows of concrete
4. Physical handling of larger pieces of concrete debris to expose underside for gamma surveying	Rough surfaces may cut and scrape skin on hands	Wear a set of work gloves to protect hands when handling concrete pieces
5. Entering Exclusion Zone (EZ) to perform survey	Tripping	Maintain good housekeeping in survey area
	Exposure to radiological contamination	PPE including booties, Tyveks, and gloves will be used
	Spread of radiological contamination outside EZ	Establish step-off area outside of EZ
6.Moving contaminated or clean material to appropriate disposal containers	Use of front end loader by untrained personnel	Ensure equipment operators are adequately trained
	Lower back strain from lifting	Proper lifting techniques will be used. Loads will be sized so as not to create unreasonable weights for manual lifting
	Exposure to radiological contamination	PPE including booties, Tyveks, and gloves will be used
	Exposure to silica	Use of a real-time dust monitor will document dust levels. Respiratory protection will be used if dust levels exceed established action levels (dependent on silica content of concrete)

7.3.6.3 Handling

The concrete rubble must be crushed to a uniform size of less than one inch to implement the disposition survey design and meet the MQOs. The crushing process will generate dust potentially containing radioactive material. Controls to limit dust generation were implemented during concrete crushing activities. Equipment involved in handling the concrete during crushing activities (e.g., front loader, crusher, rebar separator, conveyor belts, dump trucks) is categorized as impacted and will require a disposition survey before the equipment can be released. Surveys of the front loader used for these operations are discussed in Section 7.4 and Section 7.5.

7.3.6.4 Uncertainty in the Scan MDC

The two most important MQOs for this survey design are the scan MDC for the FIDLER measurements and the required measurement method uncertainty, u_{MR}, for the measurements on the systematic grid. The former will be addressed in this section, and the latter in the next. Several of the equations used in this section are discussed further in Appendices F and G.

As noted in Section 5.7.3, the MDC itself has an uncertainty which can be estimated using the methods of Section 5.6 and Appendix G.2.

From Equation F-10,

$$\text{Scan MDC} = y = C\frac{\text{MDER}}{R_T}.$$

Substituting for MDER from Equation F-9,

$$\text{MDER} = \frac{\text{MDCR}_{surveyor}}{W_T}, \text{ then}$$

$$y = C\frac{\left(\dfrac{\text{MDCR}_{surveyor}}{W_T}\right)}{R_T}.$$

Inserting Equation F-8 for $\text{MDCR}_{\text{surveyor}} = s_{i,\,surveyor} \times (60/i)$ and Equation F-7 for $s_{i,\,surveyor} = \dfrac{d'\sqrt{b_i}}{\sqrt{p}}$, we get

$$y = C\frac{\left(\dfrac{s_{i,\,surveyor}(60/i)}{W_T}\right)}{R_T} = C\frac{\left(\dfrac{\left(\dfrac{d'\sqrt{b_i}}{\sqrt{p}}\right)(60/i)}{W_T}\right)}{R_T} = \frac{60Cd'\sqrt{b_i}}{iW_TR_T\sqrt{p}} \qquad \textbf{(7-3)}$$

Where:

b_i	= the average number of counts in the background interval (214.5 counts) was chosen as a constant value in Appendix F. Here b_i will be assumed to have a triangular distribution of half-with of 30%, so the mean value of b_i will be rounded to 215 and $u(i) = 64/\sqrt{6} = 26$.
i	= the observation interval length (one second) was chosen as a constant value in Appendix F. Here i will be assumed to have a triangular distribution of half-with 0.5, so the mean value of i = 1.0 and $u(i) = 0.5/\sqrt{6} = 0.2$.
p	= efficiency of a less than ideal surveyor, range of 0.5 to 0.75 from NUREG-1507 (NRC 1998b); a value 0.5 was chosen as a conservative value in Appendix F. Here p will be assumed to have a rectangular distribution of half-with 0.125, so the mean value of p = 0.625 and $u(p) = 0.125/\sqrt{3} = 0.072$.
d'	= detectability index from Table 6.1 of NUREG-1507 (NRC 1998b); a value of 1.38 was selected, which represents a true positive detection rate of 95% and a false positive detection rate of 60%.
$s_{i,surveyor}$	= minimum detectable number of net source counts in the observation interval by a less than ideal surveyor.
$\text{MDCR}_{surveyor}$	= minimum detectable count rate by a less than ideal surveyor (cpm).
MDER	= minimum detectable exposure rate for the "*i*th" source term, by a less than ideal surveyor, (μR/h).
W_T	= total weighted instrument sensitivity (cpm per μR/h) W_T = 44,923 for natural uranium from Table F.3 and W_T = 3,881 for natural thorium from Table F.4.
R_T	= total exposure rate with buildup (μR/h) $R_T = 1.413\times10^{-4}$ for natural uranium from Table F.3 and $R_T = 2.619\times10^{-2}$ for natural thorium from Table F.4.
C	= concentration of source term (set at 1 Bq/kg in Section F.5).

Scan MDC≡ y = minimum detectable concentration by scanning (Bq/kg), where the symbol y has been introduced for the Scan MDC for simplicity of notation in the following, y_U for natural uranium and y_{Th} for natural thorium.

The uncertainties for W_T and R_T will be discussed further below.

Inserting the values above into the equation for y we obtain:

$$y_U = \frac{60Cd'\sqrt{b_i}}{iW_T R_T \sqrt{p}} = \frac{60(1)(1.38)\sqrt{215}}{(1)(44,923)(1.413\times10^{-4})\sqrt{0.625}} = 242 \text{ Bq/kg} \text{ and}$$

$$y_{Th} = \frac{60Cd'\sqrt{b_i}}{iW_T R_T \sqrt{p}} = \frac{60(1)(1.38)\sqrt{215}}{(1)(3,881)(2.619\times10^{-2})\sqrt{0.625}} = 15 \text{ Bq/kg}$$

Since we are assuming there are no correlations among the input variables, the combined standard uncertainty of y can be calculated using the following equation from Section 5.6.1:

$$u_c^2(y) = \sum_{i=1}^{N}\left(\frac{\partial y}{\partial x_i}\right)^2 u^2(x_i) = \sum_{i=1}^{N} c_i^2 u^2(x_i)\ .$$

The concentration of the source term, C, and the detectability index, d', are chosen constants, so

$$u_c^2(y) = \left(\frac{\partial y}{\partial b_i}\right)^2 u^2(b_i) + \left(\frac{\partial y}{\partial i}\right)^2 u^2(i) + \left(\frac{\partial y}{\partial p}\right)^2 u^2(p) + \left(\frac{\partial y}{\partial R_T}\right)^2 u^2(R_T) + \left(\frac{\partial y}{\partial W_T}\right)^2 u^2(W_T)$$

The sensitivity coefficients, c_i^2, are calculated as follows:

$$\left(\frac{\partial y}{\partial b_i}\right) = \frac{\partial\left(\frac{60Cd'\sqrt{b_i}}{iW_T R_T\sqrt{p}}\right)}{\partial b_i} = \left(\tfrac{1}{2}\right)\frac{60Cd'}{iW_T R_T\sqrt{p}\sqrt{b_i}} = \left(\tfrac{1}{2}\right)\frac{y}{b_i}$$

$$\left(\frac{\partial y}{\partial p}\right) = \frac{\partial\left(\frac{60Cd'\sqrt{b_i}}{iW_T R_T\sqrt{p}}\right)}{\partial p} = \left(-\tfrac{1}{2}\right)\frac{60Cd'\sqrt{b_i}}{iW_T R_T p^{3/2}} = \left(-\tfrac{1}{2}\right)\frac{y}{p}$$

$$\left(\frac{\partial y}{\partial i}\right) = \frac{\partial\left(\frac{60Cd'\sqrt{b_i}}{iW_T R_T\sqrt{p}}\right)}{\partial i} = -\frac{60Cd'\sqrt{b_i}}{i^2 W_T R_T\sqrt{p}} = -\frac{y}{i}$$

$$\left(\frac{\partial y}{\partial R_T}\right) = \frac{\partial\left(\frac{60Cd'\sqrt{b_i}}{iW_T R_T\sqrt{p}}\right)}{\partial R_T} = -\frac{60Cd'\sqrt{b_i}}{iW_T R_T^2\sqrt{p}} = -\frac{y}{R_T}$$

$$\left(\frac{\partial y}{\partial W_T}\right) = \frac{\partial\left(\frac{60Cd'\sqrt{b_i}}{iW_T R_T\sqrt{p}}\right)}{\partial W_T} = -\frac{60Cd'\sqrt{b_i}}{iW_T^2 R_T\sqrt{p}} = -\frac{y}{W_T}$$

Therefore,

$$\begin{aligned} u_c^2(y) &= \left(\frac{y}{2b_i}\right)^2 u^2(b_i) + \left(\frac{-y}{i}\right)^2 u^2(i) + \left(\frac{-y}{2p}\right)^2 u^2(p) + \left(\frac{-y}{R_T}\right)^2 u^2(R_T) + \left(\frac{-y}{W_T}\right)^2 u^2(W_T) \\ &= y^2\left[\left(\frac{u(b_i)}{2b_i}\right)^2 + \left(\frac{u(i)}{i}\right)^2 + \left(\frac{u(p)}{2p}\right)^2 + \left(\frac{u(R_T)}{R_T}\right)^2 + \left(\frac{u(W_T)}{W_T}\right)^2\right] \\ &= y^2\left[\left(\frac{26}{2(215)}\right)^2 + \left(\frac{0.2}{1}\right)^2 + \left(\frac{0.072}{2(0.625}\right)^2 + \left(\frac{u(R_T)}{R_T}\right)^2 + \left(\frac{u(W_T)}{W_T}\right)^2\right]. \end{aligned} \qquad \textbf{(7-4)}$$

The most notable sources of uncertainty associated with W_T and R_T are the modeling assumptions for the source-to-detector separation distance during scanning and the depth distribution of the radioactivity in the crushed concrete. To calculate uncertainties, the same basic modeling assumptions as those for the MDC calculations were applied, though with variations to both the source-to-detector separation distance during scanning and the distribution of the radioactivity in the crushed concrete. While the MDC calculation in Appendix F assumes a source-to-detector distance of 10 cm and that the activity is uniformly-distributed within a cylindrical volume of crushed concrete 15 cm thick with a radius of 28 cm, several other calculations were made using source-to-detector separation distances during scanning of 8, 10, and 12 cm, and by varying the distribution of the radioactivity in the crushed concrete from uniform to uniformly-distributed within both the top and bottom 7.5 cm of the cylindrical volume of crushed concrete, to assess the potential variability in the MDC. In each calculation the total activity was the same, only the distribution with depth was changed. The extreme cases were for a source-to-detector distance of 8 cm with the activity uniformly distributed within the top 7.5 cm of the concrete versus a source-to-detector distance of 12 cm with the activity uniformly distributed within the bottom 7.5 cm of the concrete. While more extreme conditions might be imagined, the foregoing were considered to represent reasonable bounds on the source-to-detector distance and the activity distribution with depth. The other assumptions used in the calculations were the same as used in Appendix F. Therefore, there are three values each to describe the distribution of the possible values of W_T and R_T : The estimated mean value calculated for a uniform distribution of radioactivity in the 15 cm of concrete surveyed at 10 cm above; an estimated lower bound calculated for a uniform distribution of radioactivity in the bottom 7.5 cm of concrete surveyed at 12 cm above; and an estimated upper bound calculated for a uniform distribution of radioactivity in the top 7.5 cm of concrete surveyed at 8 cm above.

The values for W_T and R_T at the extremes considered were not equally distant from the mean, i.e., their distribution was not symmetric. However the GUM suggests that in the absence of more information the simplest approximation is a symmetric rectangular distribution of the same total width. With this approximation, $u(W_T) = 6673$ and $u(R_T) = 4.638\times10^{-5}$ for natural uranium and $u(W_T) = 539$ and $u(R_T) = 7.315\times10^{-3}$ for natural thorium.

Using this information in Equation 7-4 we find:

$$u_c^2(y_U) = {y_U}^2\left[\left(\frac{26}{2(215)}\right)^2 + \left(\frac{0.2}{1}\right)^2 + \left(\frac{0.072}{2(0.625}\right)^2 + \left(\frac{u(R_T)}{R_T}\right)^2 + \left(\frac{u(W_T)}{W_T}\right)^2\right]$$

$$= (238)^2\left[\left(\frac{26}{2(215)}\right)^2 + \left(\frac{0.2}{1}\right)^2 + \left(\frac{0.072}{2(0.625}\right)^2 + \left(\frac{4.638\times10^{-5}}{1.413\times10^{-4}}\right)^2 + \left(\frac{6673}{44{,}923}\right)^2\right]$$

$$= 10{,}013 \text{ (Bq/kg)}^2.$$

So, with rounding,

$u_c(y_U) = 100$ Bq/kg. Therefore the FIDLER Scan MDC is $y_U = 242$ Bq/kg with an expanded uncertainty of 200 Bq/kg using a coverage factor of 2 and an estimated coverage probability of 95%. The upper bound of the Scan MDC using this interval is 442 Bq/kg.

Similarly,

$$u_c^2(y_{Th}) = y_{Th}{}^2\left[\left(\frac{26}{2(215)}\right)^2+\left(\frac{0.2}{1}\right)^2+\left(\frac{0.072}{2(0.625}\right)^2+\left(\frac{u(R_T)}{R_T}\right)^2+\left(\frac{u(W_T)}{W_T}\right)^2\right]$$

$$= (15)^2\left[\left(\frac{26}{2(215)}\right)^2+\left(\frac{0.2}{1}\right)^2+\left(\frac{0.072}{2(0.625}\right)^2+\left(\frac{7.315\times10^{-3}}{2.619\times10^{-2}}\right)^2+\left(\frac{539}{3{,}881}\right)^2\right]$$

$$= 32\ (\text{Bq/kg})^2.$$

So, with rounding,

$u_c(y_{Th}) = 6$ Bq/kg. Therefore the FIDLER Scan MDC is $y_{Th} = 15$ Bq/kg with an expanded uncertainty of 12 Bq/kg using a coverage factor of 2 and an estimated coverage probability of 95%. The upper bound of the Scan MDC using this interval is 27 Bq/kg.

The scan MDCs of approximately 438 Bq/kg for uranium and 27 Bq/kg for thorium are both less than their respective NUREG-1640-based activity action levels of 38,000 and 330 Bq/kg, respectively.

7.3.6.5 Measurement Quantifiability

MARSAME recommends the requirement $u_{MR} \leq \Delta / 10$ by default when decisions are being made about the mean of a sampled population.

For this case study, the Unity Rule, $\frac{C_{^{238}\text{U}}}{AL_{^{238}\text{U}}}+\frac{C_{^{234}\text{U}}}{AL_{^{234}\text{U}}}+\frac{C_{^{232}\text{Th}}}{AL_{^{232}\text{Th}}}+\frac{C_{^{226}\text{Ra}}}{AL_{^{226}\text{Ra}}} \leq 1$, will be used to compare the sum of the ratios of the radionuclide concentrations to their respective action levels.

In Section 7.3.5.3 the LBGR was chosen to be 0.15, so $u_{MR} \leq \Delta / 10$ = (UBGR – LBGR)/10 = (1.0 - 0.15)/10 = 0.085. Therefore the requirement on the relative uncertainty of the sum of fractions at the action level is slightly more stringent than simply requiring that the MQC be less than the action level. We require that

$$u_c\left(\frac{C_{^{238}\text{U}}}{AL_{^{238}\text{U}}}+\frac{C_{^{234}\text{U}}}{AL_{^{234}\text{U}}}+\frac{C_{^{232}\text{Th}}}{AL_{^{232}\text{Th}}}+\frac{C_{^{226}\text{Ra}}}{AL_{^{226}\text{Ra}}}\right) \leq 0.085 \text{ when } \left(\frac{C_{^{238}\text{U}}}{AL_{^{238}\text{U}}}+\frac{C_{^{234}\text{U}}}{AL_{^{234}\text{U}}}+\frac{C_{^{232}\text{Th}}}{AL_{^{232}\text{Th}}}+\frac{C_{^{226}\text{Ra}}}{AL_{^{226}\text{Ra}}}\right) = 1.0.$$

Clearly, if each of the four terms in the sum is constrained to a fourth of its limit, the unity rule will be satisfied.

If the concentrations of the radionuclides of concern are independent, then the requirement on u_c can be expressed as:

$$u_c^2\left(\frac{C_{^{238}\mathrm{U}}}{0.25AL_{^{238}\mathrm{U}}}+\frac{C_{^{234}\mathrm{U}}}{0.25AL_{^{234}\mathrm{U}}}+\frac{C_{^{232}\mathrm{Th}}}{0.25AL_{^{232}\mathrm{Th}}}+\frac{C_{^{226}\mathrm{Ra}}}{0.25AL_{^{226}\mathrm{Ra}}}\right)$$
$$=\left(\frac{u(C_{^{238}\mathrm{U}})}{0.25AL_{^{238}\mathrm{U}}}\right)^2+\left(\frac{u(C_{^{234}\mathrm{U}})}{0.25AL_{^{234}\mathrm{U}}}\right)^2+\left(\frac{u(C_{^{232}\mathrm{Th}})}{0.25AL_{^{232}\mathrm{Th}}}\right)^2+\left(\frac{u(C_{^{226}\mathrm{Ra}})}{0.25AL_{^{226}\mathrm{Ra}}}\right)^2\le\left(0.085\right)^2.$$

If the required relative measurement method uncertainty is the same for each radionuclide, then

$$\left(\frac{u(C_{^{238}\mathrm{U}})}{0.25AL_{^{238}\mathrm{U}}}\right)^2=\left(\frac{u(C_{^{234}\mathrm{U}})}{0.25AL_{^{234}\mathrm{U}}}\right)^2=\left(\frac{u(C_{^{232}\mathrm{Th}})}{0.25AL_{^{232}\mathrm{Th}}}\right)^2=\left(\frac{u(C_{^{226}\mathrm{Ra}})}{0.25AL_{^{226}\mathrm{Ra}}}\right)^2\le\left(0.085\right)^2/4=\left(0.0425\right)^2.$$

Table 7.7 Radionuclide-Specific Required Relative Measurement Method Uncertainties

Radionuclide	**Modified Action Level (Bq/kg)**	Required Relative Measurement Method Uncertainty, φ_{MR}
^{238}U	$38,000/4=9500$	4.25%
^{234}U	$12,000,000/4=3,000,000$	4.25%
^{232}Th	$330/4=82.5$	4.25%
^{226}Ra	450/4 = 112.5	4.25%

7.3.6.6 Survey Data

As the concrete is removed from the crusher, it is placed in a wooden frame (measuring 8 meters by 10 meters by 15 cm) on a concrete pad. The wooden frame's volume (12 cubic meters) corresponds to the volume associated with each sample from the survey design (i.e., 83 cubic meters divided by seven samples). Therefore, seven batches of concrete equal one survey unit. One sample is collected from the center of the concrete rubble residing in the wooden form for each batch of crushed concrete. One hundred percent of the surface is surveyed to identify locations with count rates greater than 30,000 cpm to investigate for areas of elevated activity and establish biased sampling points. A sample is collected at each location exceeding 30,000 cpm.

If no scan results exceed 30,000 cpm, the concrete is removed from the form and placed in the non-impacted concrete staging area awaiting laboratory analysis of the samples. If the scan survey identifies areas exceeding 30,000 cpm, the concrete is transferred to a holding container to control access to the concrete until the laboratory analyses are completed. A total of 126 batches of concrete are scanned (7 batches for each of the 18 survey units). Seventeen batches of concrete are segregated as potentially containing elevated levels of radioactivity based on the scan survey results, and one additional sample is collected from each batch as part of the investigation. No areas exceeding 100,000 cpm are identified during implementation of the disposition survey.

Five additional samples are collected from random locations on the floor of the administrative building to provide a total of seven reference area samples. The results of the two samples collected from the administrative building during the preliminary surveys are reviewed and determined to be of adequate quality for the disposition survey.

All of the concrete samples collected during implementation of the disposition survey are sent to a laboratory for analysis by gamma spectrometry and alpha spectrometry for uranium isotopes. Thorium-232 is quantified based on the ^{228}Ac gamma spectrometry results. Radium-226 is quantified based on the ^{214}Bi gamma spectrometry results. A total of 150 samples are analyzed, including seven samples from the reference area. The 17 biased sample locations identified by the scan survey were analyzed by gamma spectroscopy.

Performance checks of the FIDLER were made at the beginning and end of collection activities for each survey unit. These performance checks included a blank measurement in an area away from potential sources of radioactivity and a source check. Control charts were constructed to monitor the performance of the FIDLER throughout the survey. One FIDLER was dropped while performing a scan survey and the window was damaged. The instrument was removed from service and all scan measurements were repeated using a replacement FIDLER for that survey unit. No quality related problems were identified during the performance of the scan surveys.

The offsite laboratory provided the results of the laboratory analyses. The quality control measurements specified in the work plan were performed. All of the QC results were within the limits specified in the work plan. No quality related issues were identified during the performance of the sampling surveys.

7.3.7 Assess the Results of the Disposition Survey

7.3.7.1 Data Quality Assessment

The disposition survey design for the concrete rubble is verified as having been executed very closely to the survey design, with the appropriate number of measurements collected for each of the survey units.

The quality control sample results from the laboratory are reviewed and the data are deemed acceptable. An exploratory data analysis of the entire data set is performed to gain an understanding of the structure of the data.

The sum of fractions for each sample is calculated using the results for ^{238}U, ^{234}U, ^{232}Th (^{228}Ac), and ^{226}Ra (^{214}Bi) and the radionuclide specific action levels. Only two samples result in sums of fractions greater than 1.0 without correcting for background. Both of these samples came from batches that were segregated prior to crushing based on visual evidence of staining within the concrete rubble; these were also the two locations with the highest scan survey results. A frequency plot (Figure 7.1) and normal cumulative frequency plot (Figure 7.2) were constructed to provide visual representations of the data.

7.3.7.2 Wilcoxon Rank Sum Test

The Wilcoxon Rank Sum test was used to compare the reference area data to the survey unit data. In each case the test statistic exceeded the critical value of 65, so the null hypothesis was rejected for all seventeen survey units. It was concluded that the average activity in the crushed concrete exceeds background by less than a sum of fractions of 1.0.

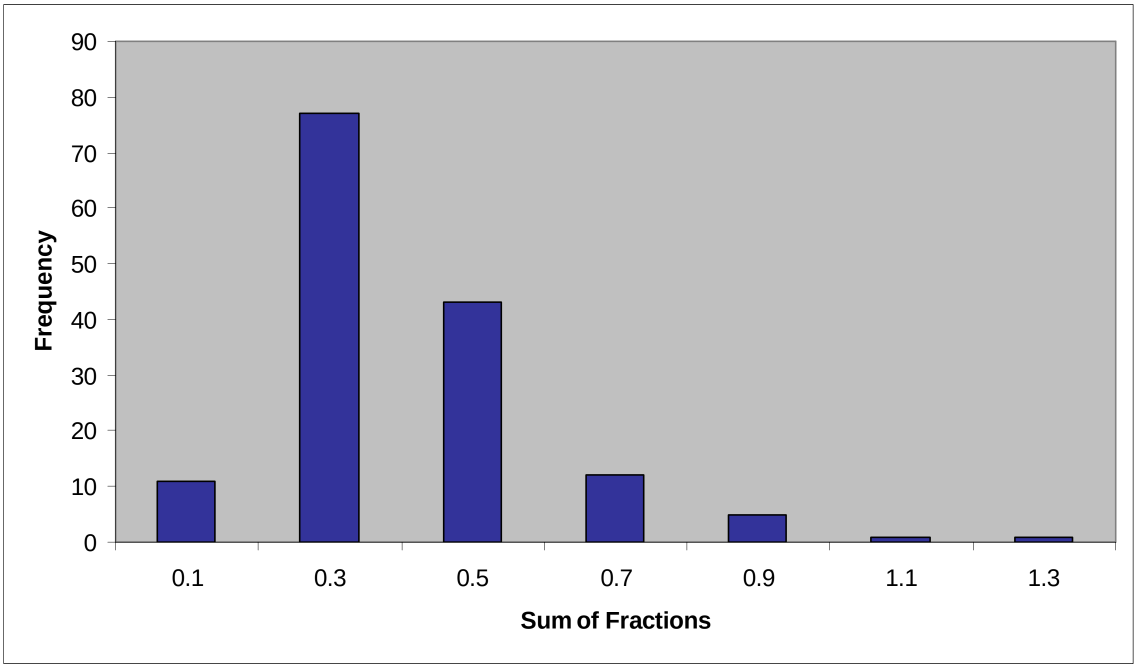

Figure 7.1 Frequency Plot of Case Study Data

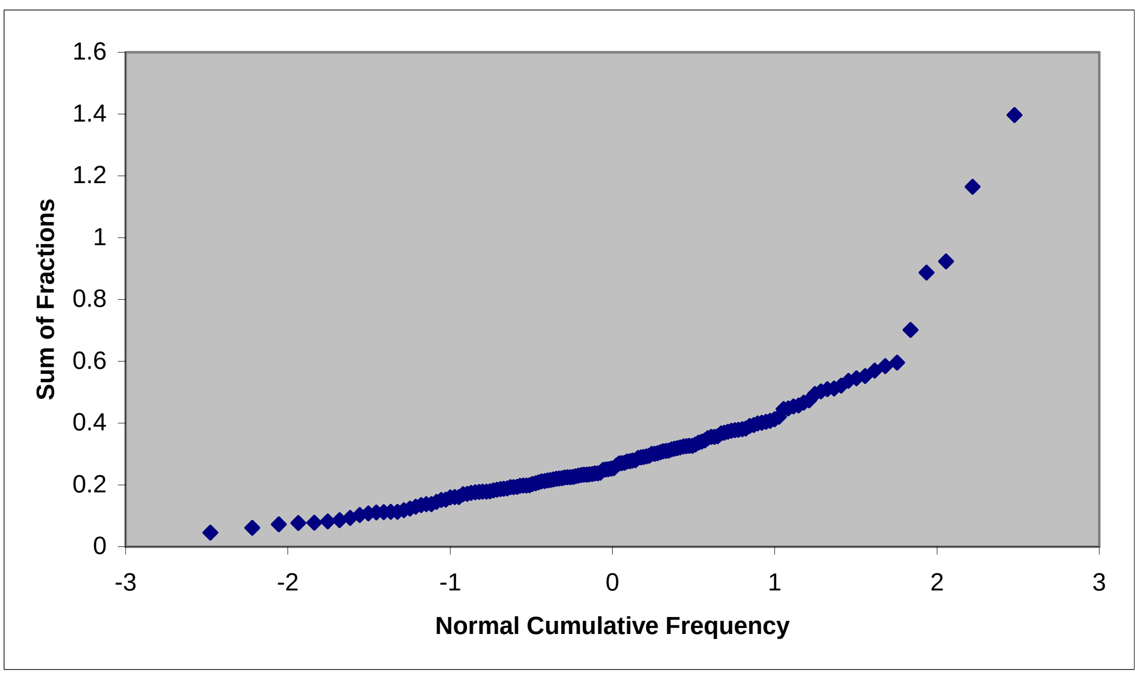

Figure 7.2 Cumulative Frequency Plot of Case Study Data

7.3.8 The Decision

In every survey unit, including those with stained concrete, the test statistic for the WRS test exceeded the critical value in Table A.4 in Appendix A. The null hypothesis that the mean sum of fractions in the survey unit exceeds 1.0 is rejected. Even though the standard deviation of the survey unit results (0.287) exceeded the variability used to design the survey (i.e., 0.15), it did not significantly impact the ability to make a decision about the concrete rubble. Based on the results of the disposition survey, the crushed concrete can be cleared.

7.4 Mineral Processing Facility Rented Equipment Baseline Survey

7.4.1 Description

Heavy equipment is required to move the piles of concrete rubble at the mineral processing facility discussed in Section 7.3. A front loader is rented to assist with the work. The radiological history of the rented front loader is unknown.

7.4.2 Objectives

The objective is to apply interdiction controls to prevent the introduction of offsite radioactive materials to the mineral processing facility. In addition, surveying the front loader before it enters the site may provide reference area data for use in clearing the front loader at the end of the project (see Section 7.5). The scope of this case study is limited to a rented front loader being brought to the site for on-site transport of impacted concrete rubble.

7.4.3 Initial Assessment of the M&E

7.4.3.1 Categorize the M&E

The material to be assessed is a rented front loader (Figure 7.3). A review of the existing information shows it is not adequate to categorize the front loader (see Figure 2.1). A visual inspection of the front loader as it is delivered to the site shows the equipment has been used, but there are no notable quantities of soil. No detailed historical records pertaining to the usage history of the front loader are available for review, other than that available from the rental company pertaining to the types of sites where heavy equipment is rented and used. Natural radionuclides are present in or commingled with soil, sediment, rubble, debris, and water. Heavy equipment is in direct contact with natural uranium and thorium during operations. Since there is a possibility the M&E may contain radionuclide concentrations or radioactivity exceeding the background at the mineral processing facility, the front loader is categorized as impacted.

Figure 7.3 Front Loader

7.4.3.2 Preliminary Surveys

The information available after categorizing the front loader is not adequate to select a disposition option (see Figure 2.2). The data gaps for the front loader are associated with describing the physical and radiological attributes of the front loader. The scoping survey design includes scanning external and easily measurable areas of the front loader that have the highest potential to contact radioactive materials.

A description of the physical attributes of the front loader is listed in Table 7.8 (per Table 2.1). The front loader is a large, complicated piece of machinery. It incorporates four wheels that are 50 centimeters (cm) (1 feet [ft], 8 inches [in]) wide and 150 cm (5 ft) tall, a wheelbase of 345 cm (11 ft, 4 in), an additional section of 246 cm (8 ft, 1 in) behind the rear wheels for the engine housing, and a height of 363 cm (11 ft, 9 in) to the top of the operator cab. The front loader uses a 320 cm–wide (10 ft, 6 in), 4.7 cubic meters–(m^3) capacity bucket (six cubic yards [yd^3]). The overall length with the bucket is 914 cm (30 ft, 0 in).

Table 7.8 Physical Attributes Used to Describe the Front Loader

Attribute	Description
Dimensions	Size - Total Mass ≈25,490 kg (56,196 lbs) Shape - Total Surface Area ≈180 m^2
Complexity	The front loader is composed of multiple materials. Most external components are painted steel. However, the tires are rubber, the cab is comprised of large sections of glass, hydraulic fluid hoses are composed of high-pressure silicon, and the joints are coated with grease. Disassembly would ideally be avoided for the considerable time and expense it adds to performing disposition surveys on the equipment. Options for surveying interior surfaces include surveying of the engine air filters and interior surfaces of the exhaust plumbing to determine whether radioactive materials have spread into the engine.
Accessibility	The inside corners of the bucket and portions of each tire and wheel are difficult to measure using conventional hand-held measurements, even with a relatively small hand-held GM detector probe. The large height of the front loader, the underside of the front loader, and the varying orientation of surfaces associated with the equipment represent a scenario that makes accessibility difficult. There are only a few porous surfaces that allow permeation of radioactivity, such as the grease used on external hinges and joints. Air inlets, grease used on external hinges and joints, and air vents in the external panels represent areas where radioactivity could penetrate to difficult-to-measure areas.
Inherent Value	The front loader can be decontaminated, reused, or recycled. The costs associated with either replacing impacted portions of the front loader, or disposing of the front loader and replacing it, are very high. As long as only exterior surfaces of the front loader become impacted, the cost of decontamination to allow unrestricted release and reuse elsewhere will probably not be substantial.

The surface area was estimated by dividing the front loader into components with regular geometric shapes and rounding to the nearest square meter. For example, the tires were modeled as cylinders and the cab was modeled as a box. The bucket has a surface area of 13.5 m^2, which is applied to the inside and outside surfaces for a total of 27 m^2. The exterior surfaces of the body have a surface area of approximately 76 m^2. The tires have a surface area of 24 m^2, and the inside of the cab is estimated at 16 m^2. Since the surfaces are not actually regular geometric shapes, a contingency factor of 25% (35 m^2) was used to account for irregular surfaces, hoses, etc. This contingency factor was based on professional judgment and approved through discussions with the regulators. The rounded total surface area is 180 m^2.

The front loader is composed of multiple materials. Most external components are painted steel. However, the tires are rubber, the cab is comprised of large sections of glass, hydraulic fluid hoses are composed of high-pressure silicon, and the joints are coated with grease. The front loader is deemed accessible, as the areas most likely to contain radioactivity are all accessible (though some portions of the front loader are more accessible than others) for conducting measurements with hand-held instruments. Internal areas of the front loader are inaccessible without disassembly.

The radiological attributes of the front loader are listed in Table 7.9 (per Table 2.2). Radionuclides of potential concern include any radionuclides that may be present. Members of the uranium and thorium radioactive decay series are used as a preliminary list of radionuclides since these are the radionuclides of concern for the site (Appendix C lists types of sites where uranium and thorium series radionuclides may be present). These are the radionuclides that are known to be present at the mineral processing facility. Radioactivity associated with the front loader is anticipated to be present at near-background concentrations. Materials may have built up in specific locations on the front loader (e.g., joints with external grease, tires, corners of the bucket) resulting in small areas of elevated radioactivity. The distribution of radioactive material is expected to be concentrated on the underside and lower edges of the front loader. Horizontal surfaces also present areas for the potential deposition of airborne radioactivity (angled and vertical surfaces also present areas for the potential deposition of airborne radioactivity but deposition of radioactivity is less likely in these areas due to surface orientation).

Given the unknown use history of the front loader, professional judgment and process knowledge are used to develop a likely scenario for the potential distribution of radioactivity. Radioactivity associated with the front loader is expected to be surficial only. Since the radioactivity is expected to be associated with materials from the site, the radioactivity is also expected to be removable. Process knowledge does not provide a likely scenario for activation or other method for volumetrically-impacting the front loader.

Table 7.9 Radiological Attributes Used to Describe the Front Loader

Attribute	Description
Radionuclides	Radionuclides of potential concern are any radionuclides that can be identified. The uranium and thorium series radionuclides are used as a preliminary list, since these are the radionuclides of concern for the mineral processing facility.
Activity	Radionuclide concentrations are expected to be close to background or zero.
Distribution	Radioactivity is expected to be associated with materials that have come in contact with the front loader. These materials will likely build up in specific locations resulting in small areas of elevated activity that can be visually identified.
Location	Radioactivity associated with the front loader is expected to be surficial and removable.

7.4.3.3 Implement Preliminary Surveys

A Geiger-Mueller (GM) meter is used to collect initial scanning survey data to help address data gaps on the bucket and tires (i.e., external and easily measurable areas of the front loader that have the highest potential for residual radioactivity). The maximum reading from the bucket was 80 counts per minute (cpm), and the maximum reading from the tires was 65 cpm. A collimated in situ gamma spectrum made of the front loader showed no gamma lines other than those associated with natural uranium, potassium, and thorium. Although one might expect some trace amounts of ^{137}Cs from atmospheric fallout, there was not enough to show up in the spectrum.

A non-impacted section of steel I-beam approximately one foot long (which resembles the majority of the surfaces of the front loader) is used as a reference material to establish the GM's background count rate. Scanning measurements are collected from flat surfaces, edges, and inside corners of the I-beam; count rates of 30 to 35 cpm are observed. Daily quality control checks were performed to ensure the instruments were operating properly.

7.4.3.4 Select a Disposition Option

The disposition options for the front loader are to accept it for use at the mineral processing facility following an interdiction survey, or to return it to the rental company.

7.4.3.5 Document the Results of the Initial Assessment

The results of the IA were documented in a letter report to the project manager. The decision to categorize the front loader as impacted was included in the report, along with the descriptions of the physical and radiological attributes of the front loader. The letter report described the scoping survey and listed the results of the measurements.

7.4.4 Identify Inputs to the Decision

Following completion of the IA, additional information needed to develop the disposition survey design is collected.

7.4.4.1 Select Radionuclides or Radiations of Concern

The initial assessment indicates that natural uranium and natural thorium are the radionuclides of potential concern.

7.4.4.2 Identify Action Levels

The action level selected for the interdiction survey is no detectable surface radioactivity above background. Since there are multiple radionuclides to be evaluated during the interdiction survey, additional discussion of action levels may be necessary.

7.4.4.3 Identify the Parameter of Interest

The parameter of interest for an interdiction survey with an action level of no detectable activity is the level of radioactivity reported for each measurement. Any measurement that detects the presence of radioactivity indicates the action level has been exceeded.

7.4.4.4 Identify Alternative Actions

The alternative actions are determined by the disposition option. If the front loader is refused access to the site, it will be returned to the rental company. If the front loader is granted access to the site, it will be used to transport concrete rubble.

7.4.4.5 Develop a Decision Rule

The decision rule incorporates the action level, parameter of interest, and alternative actions into an "if…then" statement.

If the results of any measurement identify surface radioactivity in excess of background, then the front loader will be refused access to the site. If no surface radioactivity in excess of background is detected, then the front loader will be granted access to the site.

7.4.4.6 Identify Survey Units

A survey unit is defined as the quantity of M&E for which a separate disposition decision will be made. The front loader is the survey unit. The decision rule will be applied by comparing individual measurement results to the critical value for detection. All measurements must be below the critical value (i.e., no surface radioactivity in excess of background detected) in order to accept the front loader.

7.4.4.7 Inputs for Selection of Measurement Methods

The selection of a measurement method depends on the list of radionuclides or radiations of concern and will affect the survey unit boundaries. Establishing performance characteristics for the measurement method (i.e., measurement quality objectives [MQOs]) will help ensure the measurement results are adequate to support the disposition decision.

Detection Capability

Since the action level is stated in terms of detection capability, the detection capability is critical in selecting an acceptable measurement method. The detection capability is defined as the minimum detectable concentration (MDC). The survey design will need to specify how hard to look (i.e., select an appropriate discrimination limit) before the MQO for detection capability can be established. The MDC for the selected measurement method must be less than or equal to the discrimination limit.

Measurement Method Uncertainty

The measurement method uncertainty is also important in selecting a measurement method. The MQO for detection capability will determine the acceptability of a measurement method, but it will also include information on the measurement method uncertainty. The measurement method uncertainty at background concentrations is used to calculate the MDC, as well as the critical value for the detection decision.

Range

The selected measurement method must be able to detect radionuclide concentrations or radioactivity at the discrimination limit. However, the measurement method must also be able to operate and quantify radionuclide concentrations or radioactivity at levels equal to those identified in the M&E at the site.

Specificity

The requirement for specificity will be tied to the list of radionuclides and radiations of concern. If radionuclide specific measurements are required, the measurement method must be able to identify radioactivity associated with specific radionuclides. If radionuclide specific measurements are not required, methods that measure gross activity may be acceptable.

Ruggedness

Ruggedness is not expected to be a major concern for selecting a measurement method. Since only surficial radioactivity is expected, in situ measurements of front loader surfaces will be used to collect data for comparison to the action levels. The selected measurement method must be able to perform these surface measurements in the field where the front loader is located. The environmental conditions will depend on the site location (e.g. northeast vs. southwest) and the time of the year (e.g., winter vs. summer).

7.4.4.8 Reference Materials

The majority of the surfaces on the front loader are metal (e.g., steel), although there are several rubber surfaces as well (e.g., tires, hoses). The small steel I-beam used to estimate background during the preliminary surveys will be used as the reference materials for the disposition survey. There is no inherent radioactivity from the uranium or thorium decay series expected in steel or rubber, so the selection of the reference material is not expected to result in any bias during interpretation of the results.

7.4.5 Survey Design

7.4.5.1 Select a Null Hypothesis

The hypotheses being tested are:

Null Hypothesis: The front loader contains no detectable radionuclide concentrations or radioactivity above background levels (i.e., indistinguishable from background).

Alternative Hypothesis: The front loader contains detectable radionuclide concentrations or radioactivity above background levels.

MARSAME processes require the use of Scenario B when the action level is zero, which is the case for indistinguishable from background.

7.4.5.2 Set the Discrimination Limit

The discrimination limit is the radionuclide concentration or level of radioactivity that can be reliably distinguished from the action level by performing measurements. Under Scenario B, the discrimination limit determines how hard the surveyor needs to look to determine there is no detectable radioactivity.

Acceptable surface activity levels derived from the relevant regulatory agency were selected as the discrimination limits for radionuclides of potential concern. Table 7.10 lists the potential discrimination limits based on the preliminary list of radionuclides of concern.

Table 7.10 Potential Discrimination Limits

Radionuclide of Potential Concern	Natural U	Natural Th
Average (dpm/100 cm^2)	5,000	1,000
Maximum (dpm/100 cm^2)	15,000	3,000

Based on the preliminary selection of radionuclides of potential concern, the discrimination limits for natural thorium represent the limiting case.

7.4.5.3 Limits on Decision Errors

If while scanning, an area is perceived to exceed background, a one-minute direct measurement will be performed at that location to verify the scan results. If the results of the one-minute count exceed background the front loader may not be acceptable for use on the site. Thus, there are two decisions being made for scanning surveys. The first occurs when the surveyor decides to stop and flag a location to take a direct measurement. The second is when a decision is made on whether the direct measurement exceeds background.

A Type I decision error occurs when the null hypothesis is rejected when it is true. For this survey, a Type I decision error would be refusing to allow the front loader onto the site even though there is no radioactivity present that exceeds background. The consequences of this decision error may include unnecessarily returning the front loader and taking additional time to locate a replacement, or possibly deciding to decontaminate the front loader prior to use on the site. For this reason a Type I decision error rate of 1% is specified for the direct measurements. However, during scanning the consequences of making this decision error are simply collecting additional data, so a Type I decision error rate of 60% is selected for the scanning surveys (i.e., deciding to stop and count longer when no radioactivity is present).

A Type II decision error occurs when the null hypothesis is not rejected when it is false. For this survey, a Type II decision error would be allowing the front loader to be used on the site when there is radioactivity above background. The consequences of a Type II decision error may include introducing additional radionuclides on to the site and slightly increased exposures to workers. It may also make it difficult to clear front loader and return it to the rental company when the work is complete. For this reason a Type II decision error rate of 1% is specified for the direct measurements and a Type II decision error rate of 5% is selected for the scanning.

7.4.5.4 Select a Measurement Technique

At this point in the survey design process, the planning team decides to evaluate each of the three measurement techniques to determine what might be feasible for surveying the front loader. Selection of a measurement technique will help determine the final survey design and decide between the multiple options currently available for the survey.

A scan-only survey approach requires that the measurement method be capable of detecting radioactivity at the discrimination limit. Any results exceeding the critical value would provide evidence of radioactivity levels exceeding background. There would be no need to record individual measurement results, since every result would be compared to the critical value. The calculation of the total efficiency is expected to be a major source of measurement method uncertainty. Additional measurements or assumptions are required to select a source term as the basis for the efficiency calculations. Scanning can be performed for alpha, beta, gamma, or some combination of the types of radiation. The amount of the front loader requiring scanning (i.e. 10 to 100%) would be determined by the classification. It is unknown if any scan-only measurement methods are available that meet the MQOs.

In situ survey approaches also require that the measurement method be capable of detecting radioactivity at the discrimination limit. In situ techniques allow identification of specific radionuclides, if necessary. The major source of measurement method uncertainty will likely be the model used to calculate the efficiency. Additional measurements or assumptions are required to select a source term as the basis for the efficiency calculations. The amount of the front loader requiring measurement (i.e., 10 to 100%) would be determined by the classification. The final number of measurements will be linked to the field of view of the detector. For example, a detector with a 1-m^2 field of view would require more than 180 measurements to measure 100% of the external surfaces of the front loader. An instrument such as the GM probe used during the scoping survey with a field of view of less than 100 cm^2 would require thousands of measurements to measure the minimum 10% of the front loader.

A MARSSIM-type approach would use a combination of direct measurements or samples with scanning to support a disposition decision. Sampling could damage the front loader, so direct measurements would be preferred. Locating measurements on the surface of the front loader will be problematic. Similar to scan-only and in situ designs, the scanning and direct measurements should be capable of detecting radioactivity at the discrimination limit. The MARSSIM-type survey design would require the most resources to implement.

Based on the evaluation of measurement techniques, a scan-only survey design is the preferred approach. Assumptions about the radionuclides of concern will need to be established and the availability of scan-only measurement methods needs to be verified.

7.4.5.5 Finalize Selection of Radiations to be Measured

Scan-only measurement methods are available for alpha, beta, and gamma radiations. The higher background associated with scanning for gamma radiation makes it unlikely that the measurement method could detect radioactivity at the discrimination limit. Alpha particles are attenuated more than beta particles, increasing the uncertainty caused by variations in source to detector distance. Scan-only measurement methods for beta radiation should provide the optimum survey design. However, the lower detection limits associated with alpha measurements may be required to meet the detection capability MQO. Any radioactivity in excess of background is assumed to result from natural thorium, which is the limiting radionuclide.

7.4.5.6 Develop an Operational Decision Rule

A scan-only survey will be performed for beta (and possibly alpha) radiation. Any result that exceeds the critical value associated with the MDC set at the discrimination limit will result in rejection of the null hypothesis, and the front loader will not be allowed on the site. Additional constraints on data collection activities include that the front loader be clean and dry when the measurements are performed.

7.4.5.7 Classify the M&E

The expected levels of radioactivity are background (see table 7.9). No radioactivity in excess of background is expected, so the front loader is classified as Class 3.

7.4.5.8 Select a Measurement Method

The planning team decided to verify the availability of an acceptable measurement method prior to finalizing the survey design. The GM pancake probe used to perform the scoping survey is evaluated first. The expected range of radioactivity based on the reference material and preliminary survey data is approximately 35 cpm (i.e., background) to 80 cpm.

Based on the scanning survey data collected using the GM detector during the preliminary surveys, the anticipated Scan MDC of the GM pancake may not be capable of detecting radioactivity at the discrimination limit of 1000 dpm/100 cm^2 (see Table 7.10).

An alpha-beta gas proportional detector utilizing a larger detector probe area will help achieve a lower scan MDC. The maximum reading for measurements from the bucket is 250 cpm; and the maximum reading from the tires was 220 cpm. Measurements collected from flat surfaces, edges, and inside corners of the reference material I-beam provide count rates between 180 and 190 cpm. The maximum background count rate is converted to scan MDC using NUREG 1761 (NRC 2002) equations 4-3 and 4-4.

$$s_i = d'\sqrt{b_i} = 1.38 \times \sqrt{8.3} = 4.0 \text{ counts}$$

$$\text{MDCR} = s_i \times (60/i) = 4.0 \times (60/2) = 120 \text{ cpm}$$

$$\text{Scan MDC} = \frac{\text{MDCR}}{\sqrt{p}\ \varepsilon_i \varepsilon_s} = \frac{120}{\sqrt{0.5} \times 1.29} = 132 \text{ dpm/100 cm}^2$$

Where:

b_i = the average number of background counts in the observation interval 2(250/60) = 8.3 counts)
i = the interval length (2 s) based on a scan speed of 5 cm/s
p = efficiency of a less than ideal surveyor, range of 0.5 to 0.75 from NUREG-1507 (NRC 1998b); a value 0.5 was chosen as a conservative value
d' = detectability index from Table 6.1 of NUREG-1507 (NRC 1998b); a value of 1.38 was selected, which represents a true positive detection rate of 95% and a false positive detection rate of 60%
s_i = minimum detectable number of net source counts in the observation interval (counts)
MDCR = minimum detectable count rate (cpm)
$\varepsilon_i \varepsilon_s$ = weighted total alpha-beta efficiency for natural thorium in equilibrium with its progeny on the surveyed media (1.29, see Table 7.11)

The scan MDC for activity is now below 1,000 dpm/ 100 cm^2 and is good enough to detect radioactivity at the ^{232}Th discrimination limit. However, the large size of the proportional counter may make it necessary to take some additional biased direct measurements with the GM probe in tight curves or hard to reach locations.

Table 7.11: Detector Efficiency for the Mineral Processing Facility (^{232}Th in Complete Equilibrium with its Progeny) using a Gas Proportional Detector

Radionuclide	Average Energy (keV)	Fraction	Instrument Efficiency	Surface Efficiency	Weighted Efficiency
^{232}Th	alpha	1	0.40	0.25	0.1
^{228}Ra	7.2 keV beta	1	0	0	0
^{228}Ac	377 keV beta	1	0.54	0.50	0.27
^{228}Th	alpha	1	0.40	0.25	0.1
^{224}Ra	alpha	1	0.40	0.25	0.1
^{220}Rn	alpha	1	0.40	0.25	0.1
^{216}Po	alpha	1	0.40	0.25	0.1
^{212}Pb	102 keV beta	1	0.40	0.25	0.1
^{212}Bi	770 keV beta	0.64	0.66	0.50	0.211
^{212}Bi	alpha	0.36	0.40	0.25	0.036
^{212}Po	alpha	0.64	0.40	0.25	0.064
^{208}Tl	557 keV beta	0.36	0.58	0.50	0.104
					Total efficiency = 1.29

From NUREG 1761 (NRC 2002) Table 4.3

7.4.5.9 Optimize the Disposition Survey Design

A scan-only interdiction survey will be performed of the exterior surfaces of the front loader. Since the front loader is Class 3, approximately 10% of the external surface area will be surveyed. Professional judgment will be used to select the locations for the scans in the locations with the highest potential for radioactivity (i.e., the bucket, tires, and floor of the cab). Approximately 50% of each of these areas will be surveyed, for a total of approximately 18 m^2 (7 m^2 of the bucket, 10 m^2 of the tires, and 1 m^2 of the cab floor). Experienced technicians capable of detecting radioactivity in excess of background more than 60% of the time will be used to perform the surveys. The scan speed will be 5 cm per second, so the scan should take approximately one man-hour to complete. The scans will be performed using a 100 cm^2 active probe area alpha-beta gas-proportional detector.

If while scanning, an area is perceived to exceed background, a one-minute direct measurement will be performed at that location to verify the scan results. If the results of the one-minute count exceed the critical value calculated in 7.4.6.5, the radioactivity at that location exceeds background. The results of all one-minute verification counts will be recorded on a log sheet. The location of any one-minute count that exceeds the critical value will be clearly marked.

Quality control (QC) measurements will be performed prior to the start of the survey and at the completion of the survey. These QC measurements will demonstrate that the instruments were working properly while the survey was being performed. In addition, approximately 5% of the survey will be repeated using a different surveyor to confirm the results of the initial survey.

7.4.5.10 Disposition Survey Design Documentation

The interdiction survey design was documented in a letter report to the project manager. The results of the IA were also included in this letter report.

7.4.6 Implementation of Disposition Surveys

7.4.6.1 Ensure Protection of Health and Safety

Protection of health and safety was performed as part of the survey implementation, but is not included in this case study (see Section7.3.6.1 for an example Job Safety Analysis.)

7.4.6.2 Consider Issues for Handling M&E

Since only a portion of the front loader needs to be accessed to implement the survey design, the front loader does not need to be moved to provide access to additional areas during the survey (e.g., bottom of tires, underside of bucket). Areas included in the survey do not need to be marked, outside of the small area that will be re-surveyed as part of the QC checks and locations of direct measurements exceeding the critical value. The front loader will not be parked adjacent to areas known to contain radionuclide concentrations or radioactivity in excess of background (e.g., piles of concrete rubble) while the survey is performed.

7.4.6.3 Segregate the M&E

No segregation of the front loader is required to implement the survey design.

7.4.6.4 Measurement Detection Capability

Section 7.4.4.7 established the MQO for the measurement detection capability. The scan MDC must be less than or equal to the discrimination limit.

7.4.6.5 Calculation of the Critical Value and the MDC

Both Type I and Type II errors are equally undesirable in the direct measurements made during a scan. The consequence of incorrectly alleging that the front loader is contaminated (Type I error) may raise unnecessary regulatory concerns. On the other hand, accepting a front loader that has radioactivity detectable above facility background (Type II error) may make it difficult to clear when the work is finished. Thus it is desirable to initially set $\alpha = \beta = 0.01$. The critical value for one minute static counts may be calculated from the equation in line 1 of Table 5.1:

$$S_C = z_{1-\alpha}\sqrt{N_B \frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)} = 2.326\sqrt{2\times 250} = 2.326\sqrt{500} = 52 \text{ net counts},$$

Where:

S_C is the critical value

N_B is the mean background count (250 counts)

t_S is the count time for the test source (one minute)

t_B is the count time for the background (one minute)

$z_{1-\alpha}$ is the $(1-\alpha)$-quantile of the standard normal distribution (2.326 when α =0.01).

The minimum detectable net count can be calculated from the equation in line 1 of Table 5.2:

$$\begin{aligned} S_D &= S_C + \frac{z_{1-\beta}^2}{2} + z_{1-\beta}\sqrt{\frac{z_{1-\beta}^2}{4} + S_C + N_B\frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)} \\ &= 52 + \frac{2.326^2}{2} + 2.326\sqrt{\frac{2.326^2}{4} + 52 + 250(2)} = 109 \text{ net counts}, \end{aligned}$$

Where:

$z_{1-\beta}$ is the $(1-\beta)$-quantile of the standard normal distribution (2.326 when β=0.01)

S_D is the minimum detectable value of the net instrument signal (discrimination limit, 7 cpm)

The direct measurement MDC can be calculated from equation 4-1 in NUREG 1761 (NRC 2002):

$$\text{MDC} = \frac{\text{detection limit}}{\text{total efficiency} \times \text{sample size}} = \frac{S_D}{\varepsilon_i \varepsilon_s \times \frac{\text{Probe Area}}{100}} = \frac{(109)}{(1.29) \times \frac{100}{100}} = \frac{109}{1.29} = 84.5 \text{ dpm/100 cm}^2$$

of natural thorium.

7.4.6.6 Uncertainty of the Direct Measurement MDC

$$\text{MDC} = \frac{S_D}{(\varepsilon_i \varepsilon_s) \times \frac{\text{Probe Area}}{100}}$$

Assuming a negligible uncertainty in the probe area, the combined standard uncertainty of the MDC is (see equation G-14):

$$u_c^2(\text{MDC}) = \left(\frac{\partial \text{MDC}}{\partial S_D} \right)^2 u^2(S_D) + \left(\frac{\partial \text{MDC}}{\partial \varepsilon_i \varepsilon_s} \right)^2 u^2(\varepsilon_i \varepsilon_s) \,.$$

Note that $\varepsilon_i \varepsilon_s$ is treated as a single input variable because it is the weighted total alpha-beta efficiency for natural thorium in equilibrium with its progeny on the surveyed media.

Because the MDC is of the form of a ratio of products, Equation G-15 may be used:

$$u_c^2(\text{MDC}) = \text{MDC}^2 \left(\frac{u^2(S_D)}{S_D^2} + \frac{u^2(\varepsilon_i \varepsilon_s)}{\varepsilon_i^2 \varepsilon_s^2} \right).$$

$$\begin{aligned}
S_D &= S_C + \frac{z_{1-\beta}^2}{2} + z_{1-\beta}\sqrt{\frac{z_{1-\beta}^2}{4} + S_C + N_B\frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)} \\
&= S_C + \frac{2.326^2}{2} + 2.326\sqrt{\frac{2.326^2}{4} + S_C + N_B(2)} \\
&= z_{1-\alpha}\sqrt{N_B\frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)} + \frac{2.326^2}{2} + 2.326\sqrt{\frac{2.326^2}{4} + \left(z_{1-\alpha}\sqrt{N_B\frac{t_S}{t_B}\left(1+\frac{t_S}{t_B}\right)}\right) + 2N_B} \\
&= 2.326\sqrt{N_B(2)} + \frac{2.326^2}{2} + 2.326\sqrt{\frac{2.326^2}{4} + \left(2.326\sqrt{N_B(2)}\right) + 2N_B}
\end{aligned}$$

Where the formula for S_C and the values of the constants have been inserted. The uncertainty in the time is assumed negligible, so these have also been treated as constants. Thus, the uncertainty in S_D will be due entirely to the uncertainty in the background count:

$$u^2(S_D) = \left(\frac{\partial S_D}{\partial N_B}\right)^2 u^2(N_B)$$

The sensitivity coefficient for S_D at $N_B = 250$ is

$$\left(\frac{\partial S_D}{\partial N_B}\right) = \left(\frac{\partial\left(2.326\sqrt{N_B(2)} + \frac{2.326^2}{2} + 2.326\sqrt{\frac{2.326^2}{4} + \left(2.326\sqrt{N_B(2)}\right) + 2N_B}\right)}{\partial N_B}\right)$$

$$= \left(\frac{\partial\left(2.326\sqrt{N_B(2)}\right)}{\partial N_B}\right) + \frac{\partial\left(\frac{2.326^2}{2}\right)}{\partial N_B} + \frac{\partial\left(2.326\sqrt{\frac{2.326^2}{4} + \left(2.326\sqrt{N_B(2)}\right) + 2N_B}\right)}{\partial N_B}$$

$$= \left(\frac{\left(2.326\sqrt{2}\right)}{2\sqrt{N_B}}\right) + 0 + 2.326\left[\frac{\partial\left(\sqrt{\frac{2.326^2}{4} + \left(2.326\sqrt{N_B(2)}\right) + 2N_B}\right)}{\partial\left(\frac{2.326^2}{4} + \left(2.326\sqrt{N_B(2)}\right) + 2N_B\right)}\right]\left[\frac{\partial\left(\frac{2.326^2}{4} + \left(2.326\sqrt{N_B(2)}\right) + 2N_B\right)}{\partial N_B}\right]$$

$$= \left(\frac{1.6447}{\sqrt{N_B}} \right) + 2.326 \left[\frac{\partial \left(\sqrt{0.5815 + 3.289\sqrt{N_B} + 2N_B} \right)}{\partial \left(0.5815 + 3.289\sqrt{N_B} + 2N_B \right)} \right] \left[\frac{\partial \left(0.5815 + 3.289\sqrt{N_B} + 2N_B \right)}{\partial N_B} \right]$$

$$= \frac{1.6447}{\sqrt{N_B}} + \frac{1.163\left(\frac{1.6447}{\sqrt{N_B}} + 2 \right)}{\sqrt{\left(0.5815 + 3.289\sqrt{N_B} + 2N_B \right)}}$$

$$= 0.104 + \frac{2.447}{23.5} = 0.208$$

Suppose the spatial variability in N_B can be described by a triangular distribution with mean 250 with a half-width of 50, then,

$u(N_B) = 50/\sqrt{6} = 20.4$.

Then $u(S_D) = \left(\frac{\partial S_D}{\partial N_B} \right) u(N_B) = (0.208)(20.4) = 4.2$.

A complete analysis of the uncertainty in, $\varepsilon_i\varepsilon_s$, the weighted total alpha-beta efficiency for natural thorium in equilibrium with its progeny on the surveyed media would involve a propagation of uncertainty through all if the input quantities in Table 7.11. A simpler estimate may be made by assuming that if all instrument efficiencies could be in error by as much as 10% and all the surface efficiencies could be in error by as much as 30%, both in the same direction, then the uncertainty in $\varepsilon_i\varepsilon_s$ might be roughly estimated with a triangular distribution with a half-width of 0.5. A triangular distribution is used because values near the mean are considered to be more likely than those at the extremes. Then, $u(\varepsilon_i\varepsilon_s) = 0.5/\sqrt{6} = 0.20$.

Putting this information together into the equation for the combined total variance of the MDC, given earlier in this section, we have:

$$
\begin{aligned}
u_c^2(\mathrm{MDC}) &= \mathrm{MDC}^2\left(\frac{u^2(S_D)}{S_D^2}+\frac{u^2(\varepsilon_i\varepsilon_s)}{\varepsilon_i^2\varepsilon_s^2}\right)\\
&= 84.5^2\left(\frac{4.2^2}{109^2}+\frac{0.20^2}{1.29^2}\right)\\
&= 7{,}140\left(0.000148+.024\right)\\
&= 172.4
\end{aligned}
$$

So the estimated combined standard uncertainty in the MDC is $u_c(\mathrm{MDC}) = 13.1$.

7.4.6.7 Quality Control

The required QC measurements are performed as described in the survey design.

7.4.6.8 Survey Data

Data from the survey of the front loader is collected consistent with the survey design and provides a complete record of the data collected. Thirty-seven locations were flagged during the survey for investigations using direct measurements. None of the direct measurement results exceeded the critical value.

7.4.7 Assess the Results of the Disposition Survey

7.4.7.1 Data Quality Assessment

The surveying procedure utilized for the front loader was verified as having been executed very closely to the survey design, with the appropriate survey coverage. The results of the QC measurements demonstrated that the instruments were working properly and a different surveyor could duplicate the results of the survey. Control charts used to check the performance of the survey instruments did not identify any potential problems with the instruments.

7.4.7.2 Preliminary Data Review

The preliminary data review for this baseline survey does not yield identifying patterns, relationships, or potential anomalies. The locations of the additional investigations appear to be randomly located based on visual inspection of the front loader.

7.4.7.3 Statistical Tests

The statistical test selected for this in situ survey is direct comparison to the critical level. If all the results are below the critical level associated with the discrimination limit, there is no detectable radioactivity above background. All of the scanning results that exceeded the critical value were subjected to additional investigation. All of the results of the additional investigations were below the critical value.

7.4.8 The Decision

Based on the results of the baseline survey, the front loader is determined to have no detectable radioactivity above background and is therefore allowed to enter the site.

7.5 Mineral Processing Facility Rented Equipment Disposition Survey

7.5.1 Description

The radiological surveys at the mineral processing facility described in Section 7.3 have been completed. The front loader that was brought on site to assist with handling the concrete rubble (see Section 7.4) is no longer being used. The front loader must be cleared before it can be returned to the rental company.

7.5.2 Objectives

The objective is to demonstrate the front loader can be cleared. The scope of this case study is limited to the rented front loader used for the on-site transport of impacted concrete rubble.

An interdiction survey was performed to demonstrate there was no detectable radioactivity associated with the front loader when it entered the site. This case study provides a comparison between interdiction and clearance surveys performed on the same piece of equipment.

7.5.3 Initial Assessment of the M&E

7.5.3.1 Categorize the M&E

The existing information is adequate to categorize the front loader. The front loader was used to transport concrete rubble containing radionuclides with concentrations exceeding background. The front loader is impacted. Following use, the front loader was steam cleaned to remove loose dirt and grease (together with any associated radioactivity) for acceptance by the rental company.

7.5.3.2 Design and Implement Preliminary Surveys

The description of the physical attributes associated with the front loader has not changed (see Table 7.7). The uranium series and thorium series radionuclides listed in Table 7.2 are the radionuclides of potential concern for the front loader. The existing information is adequate to select a disposition option, and there are no data gaps.

7.5.3.3 Select a Disposition Option

The preferred disposition option for the front loader is clearance. The existing interdiction survey design used to allow the front loader access to the site will be evaluated for applicability as a clearance survey (see Section 7.5.4.2).

7.5.3.4 Document the Results of the Initial Assessment

The decision to categorize the front loader as impacted will be documented with the results of the survey. The planning team determined that no other documentation is necessary.

7.5.4 Inputs to the Decision

7.5.4.1 Action Levels

The action level selected for the interdiction survey was no detectable surface radioactivity above background. The action levels in this case are the limits shown in Table 7.10. The limiting value is 1000 dpm/100 cm^2 for natural thorium.

7.5.4.2 Evaluate an Existing Survey Design

Since the same front loader is being surveyed, the measurement methods are still adequate. The scan MDC of 132 dpm/100 cm^2 for natural thorium is well below the action level. There were no problems identified during the interdiction survey that would prevent using the measurement methods for a clearance survey. The population parameter of interest and the survey unit boundaries are linked to the measurement method (see Sections 7.4.4.3 and 7.4.4.6).

The alternative actions are different for the clearance survey. If the front loader is cleared, it will be returned to the rental company. If the front loader is not cleared, it will remain on site. This results in a change to the decision rule. If the results of any measurement identify surface radioactivity in excess of background, the front loader will remain on site and radiological controls will remain in place. If no surface radioactivity in excess of 1,000 dpm/100 cm^2 over background is detected, the front loader will be cleared and returned to the rental company.

7.5.5 Survey Design

7.5.5.1 Select the Null Hypothesis

The hypotheses being tested are:

Null Hypothesis: The front loader contains detectable radionuclide concentrations or radioactivity in excess of 1,000 dpm/100 cm^2 above background levels

Alternative Hypothesis: The front loader contains radionuclide concentrations or radioactivity less than 1,000 dpm/100 cm^2 above background levels.

Since the action level is not zero, Scenario A is being used.

7.5.5.2 Set the Discrimination Limit

The discrimination limit is the radionuclide concentration or level of radioactivity that can be reliably distinguished from the action level by performing measurements. Under Scenario A, the discrimination limit should represent a prudently conservative estimate of any amount of natural thorium that may be present on the front loader in excess of background.

7.5.5.3 Specify Limits on Decision Errors

A Type I decision error occurs when the null hypothesis is rejected when it is true. For this survey, a Type I decision error would be clearing the front loader when there is radioactivity detectable more than 1,000 dpm/100 cm^2 above background. The consequences of a Type I decision error may include releasing radionuclides from the site and increased exposures to members of the public.

A Type II decision error occurs when the null hypothesis is not rejected when it is false. For this survey, a Type II decision error would be refusing to clear the front loader even though the radioactivity present exceeds background by less than 1,000 dpm/100 cm^2. The consequences of this decision error may include unnecessarily remediating the front loader, incurring additional costs for extra rental time, or even purchasing the front loader and disposing of it as low-level radioactive waste.

The existing survey design minimizes the potential for Type II decision errors by performing additional direct measurements when scanning results are perceived to exceed background and also by having experienced technicians perform the survey.

7.5.5.4 Classify the M&E

The potential for radioactivity exceeding background has increased since the front loader is known to have contacted concrete rubble containing radionuclides at concentrations that exceed background. This increased potential for radioactivity exceeding background results in a higher classification for portions of the front loader for the clearance survey. The inside of the bucket is now classified as Class 1. The remaining external surfaces are considered Class 3 so professional judgment can still be used to determine where surveys will be performed.

7.5.5.5 Optimize the Existing Survey Design

The front loader will be scanned with an alpha-beta gas proportional detector. Experienced technicians will perform the surveys. If while scanning, an area is perceived to exceed background, a one-minute direct measurement will be performed at that location to verify the scan results. If the results of the one-minute count exceed 1,000 dpm/100 cm^2 above background the front loader will require further remediation before it can be released. The results of all one-minute verification counts will be recorded on a log sheet. The location of any one-minute count that exceeds the critical value will be clearly marked.

Based on the classification of the inside of the bucket as Class 1, 100% of the inside of the bucket will be surveyed. In addition, 25% of the outside surface of the bucket will be surveyed, concentrating on the bottom where the bucket frequently came in contact with the concrete rubble. Similar to the interdiction survey, 50% of the tires and the floor of the cab will be surveyed. In addition, 10% of the bottom and 5% the top (i.e., horizontal surfaces) will be included in the clearance survey. Areas to be scanned will be biased to locations where residual dirt or grease is visible. The increased surface area to be scanned is expected to increase the scan time to approximately three man-hours. Based on professional judgment, four times as many investigations are expected for the clearance survey, or approximately 150 one-minute direct measurements. The additional investigations are expected to require an additional three man-hours.

Implementation of this survey design will likely identify locations with radioactivity levels exceeding 1,000 dpm/100 cm^2 above background. To minimize these occurrences, the front loader will be steam cleaned and dried prior to implementing the survey design. Locations on the bucket (which is a Class 1 survey unit) where the additional direct measurement exceeds the action level will be delineated using scanning techniques, scrubbed clean to remove any surface radioactivity, and re-surveyed (i.e., clean-as-you-go).

7.5.5.6 Disposition Survey Design Documentation

The modified survey design was documented in a letter report to the project manager. The letter report included the results of the categorization decision (see Section 7.5.3.1).

7.5.6 Implementation of Disposition Surveys

The front loader was positioned on a concrete pad during steam cleaning operations. The water was collected and containerized for survey prior to release. The bucket was lifted off the ground and supported with wooden beams to provide access to the bottom of the bucket.

The survey was implemented as described in the survey design. The beta background in the area underneath the bucket was higher than expected (i.e., 350 cpm instead of the 250 cpm used to design the survey). The bucket was lifted higher off the ground (i.e, 1.5 meters instead of 15 cm) and the scan survey was repeated with a lower background.

The survey results were documented in a letter report to the project manager.

7.5.7 Assess the Results of the Disposition Survey

7.5.7.1 Data Quality Assessment

The surveying procedure utilized for the front loader was verified as having been executed very closely to the survey design. The surveys included the appropriate scan coverage and number of additional investigations. The preliminary data review for this baseline survey does not yield identifying patterns, relationships, or potential anomalies. Control charts documenting the results of quantitative QC checks and performance checks indicate the DQOs have been achieved for this clearance survey.

7.5.7.2 Statistical Tests

The statistical test selected for this in situ survey is direct comparison to the action level of 1,000 dpm/100 cm^2 above background. If all of the measurement results are below the action level, the average natural thorium above background cannot exceed 1,000 dpm/100 cm^2 above background.

At 83 locations the scan MDC of 132 dpm/100 cm^2 above background appeared to be exceeded. However, none of the one-minute follow up counts at those locations exceeded 500 dpm/100 cm^2 above background.

7.5.8 The Decision

Based on the results of the disposition survey, the front loader is determined to have no radioactivity above the action level and so can be cleared.